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9.2-1	Station Service Water System (M1-0233, M1-0233-A, M1-0234, M2-0233, M2-0233-A, M2-0234) Chlorination System (M1-0240)
9.2-2	Deleted
9.2-3	Component Cooling Water System (M1-0229, M1-0229-A, M1-0229-B, M1-0230, M1-0230-A, M1-0230-B,

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9.2-4	Water Treatment (M1-0224, M1-0224-A)
9.2-4A	Surface Water Pre-Treatment System (M1-0224-01, M1-0224-01A, M1-0224-01B, M1-0224-01C, M1-0224-01D, M1-0224-01E, M1-0224-01F, M1-0224-01G, M1-0224-01H, M1-0224-01J, M1-0224-01K, M1-0224-01L, M1-0224-01M, M1-0224-01N)
9.2-5	Demineralized and Reactor Make Up Water (M1-0241, M1-0241-A, M1-0241-01, M1-0242, M1-0242-A, M1-0242-B, M2-0241, M2-0242)
9.2-6	Potable and Sanitary Water System (M1-0227, M1-0227-01, M1-0227-03)
9.2-7	Sewage Treatment System (M1-0226)
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9.3-2	Deleted
9.3-3	Deleted
9.3-4	Process Sampling System (M1-0228, M1-0228-A, M1-0228-B, M1-0228-C, M1-0228-01, M1-0228-02, M2-0228, M2-0228-A, M2-0228-B, M2-0228-C, M2-0228-01)
9.3-5	Vents and Drains System Containment Building (M1-0238, M1-0238-A, M2-0238, M2-0238-A)
9.3-6	Vents and Drains System Safeguards and Auxiliary Building (M1-0236, M1-0236-A, M1-0236-B, M2-0236, M2-0236-A, M2-0236-B)
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9.3-8	Vents and Drains System Turbine and Fuel Handling Building (M1-0237, M1-0237-A, M2-0237, M2-0237-A)
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9.3-10	Chemical and Volume Control System (M1-0253, M1-0253-A, M1-0254, M1-0255, M1-0255-01, M1-0255-02, M1-0256, M1-0256-A, M1-0257, M2-0253, M2-0253-A, M2-0254, M2-0255, M2-0255-01, M2-0255-02, M2-0256-A, M2-0256-B)
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9.4-5	Ventilation Containment (M1-0300, M1-0300-A, M2-0300, M2-0300-A)
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9.4-10	Non-ESF Charcoal Filter Train
9.4-11	Ventilation Chilled Water System (M1-0307, M1-0307-A, M1-0307-B, M1-0307-C, M2-0307, M2-0307-A)
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9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

9.1.1.1 Design Bases

New fuel is stored in racks (Figure 9.1-1) composed of individual vertical cells fastened together in any number to form a module which can be firmly bolted to anchors in the floor of the new fuel storage pit. The new fuel storage racks are designed to include storage for two thirds core at a center-to-center spacing of 21 inches. If the new fuel assemblies are stored dry, this spacing provides a minimum separation between adjacent fuel assemblies of 12 in., which is sufficient to maintain a subcritical array as described in Section 4.3.2.6. All surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel.

The racks are designed to withstand normal operating loads as well as Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) seismic loads meeting ANS Safety Class 3 [12] and ASME B&PV Code, Section III, Appendix XVII requirements. The new fuel racks are designed to withstand a maximum uplift force of 5000 lb.

9.1.1.2 Facilities Description

Both units of the CPNPP are serviced by a common Fuel Building which houses facilities for the storage and transfer of new and spent fuel. The Fuel Building is a controlled leakage building designed to seismic Category I requirements. For a description of the structural design considerations, see Section 3.8. The ventilation system is discussed in Section 9.4.4. The locations of the fuel storage areas within the station complex are shown on plan and elevation drawings; see Section 1.2 and Figures 1.2-38 through 1.2-40. The fuel storage and handling facilities are built in accordance with NRC Regulatory Guide 1.13.

New fuel assemblies are delivered to the site in United States Department of Transportation (DOT) approved containers. The containers are brought into the new fuel receiving area by the Fuel Building crane. Here a container is opened and the assemblies are unloaded and inspected.

Once the inspection is completed, the new fuel assembly is inserted in the new fuel storage rack (see Figure 9.1-1). The protective cover on each fuel assembly must be removed from the fuel assembly or must be open at the bottom so that water will not collect in the protective cover.

New fuel assemblies and control rods are stored in a reinforced concrete pit located in the Fuel Building. The pit, an integral part of the Fuel Building, is provided for temporary dry storage and is equipped with storage racks of sufficient capacity for approximately one-third core for each unit (total 132 fuel assemblies).

All surfaces that come into contact with fuel assemblies are made of austenitic stainless steel, thus precluding significant materials compatibility problems.

For the structural design considerations, including the loading criteria (loading and load combinations) for the Fuel Building, see Section 3.8.

The probability of a dropped mass damaging a new fuel assembly is very remote, for the following reasons:

1. New fuel racks located in the new fuel pit area are protected from dropped objects by a protective steel cover.

2. Administrative controls or interlocks, or both are used to prevent the handling of loads heavier than a fuel assembly and the associated handling tools over the new fuel storage area.

3. Safe handling features of the new fuel assembly handling tool are discussed in Subsection 9.1.4.2.3.

In preparation for refueling, the individual fuel assemblies are transported from the new fuel storage racks to the new fuel elevator using the fuel handling bridge crane or the Fuel Building overhead crane equipped with the new fuel handling tool. When an assembly has been lowered by the elevator, the fuel handling bridge crane equipped with the spent fuel handling tool can be used to place it either in the spent fuel pool for interim storage or in the Fuel Transfer System fuel basket for immediate transport into the Containment. For additional information on the fuel handling system, see Subsection 9.1.4.2.

The manipulation of new fuel assemblies will be performed by personnel trained in proper fuel handling techniques and, in addition, will use fuel handling procedures which contain provisions to assure that damage to fuel assemblies during movement is prevented.

Details of the seismic design and testing of the new fuel storage area are presented in Section 3.7B.

For general arrangement of new fuel storage facilities, see Section 1.2 and Figures 1.2-38, 1.2-39, and 1.2-40.

9.1.1.3 Safety Evaluation

The design of normally dry new fuel storage racks is such that the effective multiplication factor (keff) does not exceed the criteria described in Section 4.3.2.6. Consideration is given to the inherent neutron absorbing effect of the materials of construction. The detailed criticality safety evaluation is discussed in Section 4.3.2.6.

The design of the fuel storage rack assembly is such that it is impossible to insert the new fuel assemblies in other than prescribed locations, thereby preventing any possibility of accidental criticality.

The fuel storage racks are designed to withstand shipping, handling, and normal operating loads (dead loads of fuel assemblies), as well as SSE loads; these racks meet ANS Safety Class 3 requirements. The fuel storage racks are also designed to meet the seismic Category I requirements of NRC Regulatory Guide 1.29, as discussed in Section 1A(B).

The potential for inadvertent criticality has been evaluated for accidents that could result in alteration of the geometric spacing of fuel assemblies. For the normally dry conditions of the new fuel storage racks, fuel with an enrichment of 5 w/o U235 or less cannot be made critical, even

considering any damage or deformation that could occur as a result of dropped heavy loads or crane uplift forces on the racks. Further, it is not necessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, wet moderated conditions need not be considered coincident with accidental deformation or damage to the normally dry new fuel storage racks. It is concluded that such accidents would not results in advertent criticality.

Shielding requirements are discussed in Subsection 9.1.4.3.4.

Design of this storage facility is in accordance with NRC Regulatory Guide 1.13, Revision 1, December 1975, ensuring a safe condition under normal and postulated accident conditions.

9.1.2 SPENT FUEL STORAGE IN THE SPENT FUEL POOLS AND CONTAINMENTS

9.1.2.1 Design Bases

High density spent fuel storage is provided in the two common pools, Spent Fuel Pool 1 (SFP1) and Spent Fuel Pool 2 (SFP2), and low density spent fuel storage is provided inside each reactor containment. The storage in each spent fuel pool is comprised in two regions: Region I high density storage racks and Region II high density storage racks. The criticality design basis for Region I and II racks is provided in Section 4.3.2.6. Region I racks are primarily designed for refueling operations and the Region II racks are designed for long term storage. The design bases for the Region I and Region II racks is contained in Reference 18. The low density storage inside containment is designed for temporary storage in support of refueling operations.

The racks maintain a separation between spent fuel assemblies sufficient to maintain subcriticality as described in Section 3.1.6.3 and Section 4.3.2.6. All surfaces that come into contact with fuel assemblies are made of annealed austenitic stainless steel which is resistant to corrosion during normal and emergency water quality conditions.

Spent fuel storage racks are designed to withstand shipping, handling, normal operating loads (dead loads of fuel assemblies), as well as SSE loads; these racks meet ANS Safety Class 3 and ASME B&PV Code, Section III, Appendix XVII requirements. The spent fuel storage racks are also designed to meet the seismic Category I requirements of Reg. Guide 1.29, Revision 2, February 1976.

The spent fuel storage racks have adequate energy absorption capabilities to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the fuel handling bridge crane. Cranes capable of carrying loads heavier than a spent fuel assembly are prevented by interlocks or administrative controls, or both, from traveling over the spent fuel storage areas when fuel is stored in them.

The spent fuel storage racks can withstand an uplift force equal to the uplift force of the spent fuel pool bridge hoist.

Shielding requirements are discussed in Subsection 9.1.4.3.4.

9.1.2.2 Facilities Description

Two pools are provided for CPNPP spent fuel storage. Spent fuel assemblies and irradiated control rods are stored underwater in racks after transfer from the reactor. The fuel assemblies and control rods are held vertically in the racks located on the floor of the spent fuel storage pools. The two reinforced concrete pools are stainless- steel lined and are an integral part of the Fuel Building. For the structural design considerations of the Fuel Building, including the loading criteria, see Section 3.8. The spent fuel racks are designed to accommodate an SSE, shipping, and handling loads, and the dead load of the spent fuel assemblies.

The spent fuel assemblies in SFP1 are stored in high density Region I and Region II racks with a usable capacity of 1,684 storage cells. The spent fuel assemblies in SFP2 are stored in high density Region I and Region II racks with a usable capacity of 1,689 storage cells. This provides a total storage space for the two pools of 3,373 fuel assemblies.

The high density Region I rack (Figure 9.1-2B) is composed of vertical cells fastened together in a checkerboard arrangement to produce a matrix structure. The cells are welded to a baseplate and to one another to form an integral structure without the use of a supporting grid structure. The high density Region I racks in Spent Fuel Pool 1 (SFP1) and Spent Fuel Pool 2 (SFP2) are free standing and self supporting. The center to center spacing between cells within a Region I rack is a nominal 10.65 inch by a nominal 11.05 inch. The Region I racks use a flux trap design and have neutron absorbing “Boral” panels between adjacent storage cells to provide neutron attenuation.

The high density Region II rack (Figure 9.1-2A) is composed of vertical cells fastened together in a checkerboard arrangement to produce a matrix structure. The cells are welded to a baseplate and to one another to form an integral structure without the use of a supporting grid structure. The high density Region II racks in SFP1 and SFP2 are free standing and self supporting. The center to center spacing between cells within a Region II rack is a nominal 9.0 inches. The Region II racks do not use a flux trap design and have no special neutron absorbing material.

SFP1 contains two (2) 10 x 8 Region I rack modules, one (1) 9 x 8 Region I rack module, six (6) 12 x 14 Region II rack modules, and three (3) 11 x 14 Region II rack modules (twelve racks total). Some of the Region I cells have been modified at the top to prevent interference with the spent fuel pool swing gate. Some of the Region II cells in SFP1 have been modified to allow for fuel inspection.

SFP2 contains two (2) 10 x 8 Region I rack modules, one (1) 9 x 8 Region I rack module, six (6) 12 x 14 Region II rack modules, and three (3) 11 x 14 Region II rack modules (twelve racks total). Some of the Region I cells in SFP2 have been modified to allow for fuel inspection. Some of the Region I cells in SFP2 have been modified at the top to prevent interference with the spent fuel pool swing gate.

The containment refueling cavity of each unit has additional interim storage space for one (1) low density 5 x 5 rack module. Each low density rack is composed of individual vertical cells fastened together within a grid structure to form a module (Figure 9.1-2). The low density rack in the containment refueling cavity is firmly bolted to the floor of the cavity. The center to center spacing between cells within the containment refueling cavity storage rack is a nominal 16.0 inches.

See Section 9.1.2.1 for description of the design basis. See Section 3.1.6.3 and Section 4.3.2.6 for a more detailed discussion of prevention of criticality and storage patterns.

Each spent fuel pool is designed to safely store the irradiated fuel assemblies. A separate pit is provided as a loading area for the spent fuel shipping cask. The refueling cavities, spent fuel pools, and cask pit are connected with a common transfer canal. Each connection between the transfer canal and the spent fuel pools can be closed by using gates (see Section 9.1.4.2.3 and Figure 1.2-39).

The spent fuel pools, transfer canal, and cask pit are lined with stainless steel plate.

The reactor is refueled using equipment that handles the spent fuel assemblies underwater from the time they leave the reactor vessel until they are placed in a cask for shipment from the site.

The Fuel Building fuel handling bridge crane, provided for spent fuel handling, has a wheel mounted walkway which spans spent fuel pools, the transfer canal, and cask pit. The bridge carries an electric monorail hoist on an overhead structure which is provided with an antiderailing device and is designed to withstand an SSE. The fuel assemblies are moved within a spent fuel pool by means of a long-handled tool suspended from the hoist. For general arrangement of spent fuel storage facilities, see Section 1.2, Figures 1.2-12, 1.2-13, 1.2-15, 1.2-18, 1.2-19, 1.2-38, 1.2-39, and 1.2-40.

The manipulation of spent fuel assemblies will be performed by personnel trained in proper fuel handling techniques and, in addition, will use fuel handling procedures which contain provisions to assure that damage to fuel assemblies during movement is prevented.

Once the fuel is stored in the spent fuel pool, the Spent Fuel Pool Cooling and Cleanup System ensures continuous cooling. (See Subsection 9.1.3.3.) There are no drains or permanently connected systems or other features that can cause a loss of coolant that would uncover fuel.

Normal makeup water, to compensate evaporation losses, is supplied from the reactor makeup water system or the demineralized water supply system. In the case of a failure or malfunction of the demineralized water supply, the safety-related (seismic Category I, Safety Class 3, and redundant) Reactor Makeup Water System supplies reactor coolant purity water to the spent fuel pools. For a detailed discussion, see Section 9.2.3. Water level monitoring equipment is discussed in Subsection 9.1.3.

To limit the dose rate at the surface of the pools to 2.5 mR/hr a minimum water shielding depth of 10 ft is provided above a fuel assembly. The design low-water level provides a positive margin to the minimum water shielding requirement with a fuel element located above the spent fuel storage racks during fuel handling operations. The maximum height to which the fuel elements can be lifted is limited by the design of the hoist and the spent fuel handling tool controls. For a detailed description of shielding design, see Section 12.1. Consideration of criticality safety analysis is discussed in Section 3.1.6.3, Section 4.3.2.6 and Subsection 9.1.2.3.

Details of seismic design and testing are presented in Section 3.7B.

Sealed bearings or other measures, such as protective pans, are used to prevent the lubricant of the cranes from contaminating the spent fuel pools. The crane control and power systems are

capable of permitting continuous operation at minimum speed or frequent jogging without detrimental effects on any circuit or component.

Either spent fuel pool can be used for storage of fuel assemblies from both reactors as there are no adverse implications of sharing. In fact, sharing between the two pools permits greater flexibility.

The fuel storage facilities are designed in accordance with NRC Regulatory Guide 1.13.

When fuel assembly decay heat has reached an acceptable level, the fuel assembly can be removed from the spent fuel pool and loaded into a spent fuel shipping cask.

The following design features of the Fuel Building Overhead Crane are provided in order to prevent a cask from dropping:

1. The crane is designed to the requirements of seismic Category I. As such it can retain the maximum design load during a SSE and remain in place under all postulated seismic loadings.

2. To preclude any swinging or pendulum action of the block upon failure of one system, each wire rope system is reeved to both sides of the bottom block and upper block system.

The Fuel Building Overhead Crane is prevented by interlocks from moving over the new fuel pit during cask handling operations. The maximum lifting height for a loaded spent fuel cask is less than 30 feet. Mechanical antiderailing devices which prevent crane from being dislodged from the rail due to horizontal and vertical motion during an earthquake are provided on the Fuel Building Overhead Crane and designed to withstand an SSE. The concrete floors can withstand a fully loaded cask drop from the maximum lifting height of 29.25 feet.

A more detailed description of the Fuel Building Overhead Crane is provided in Section 9.1.4.

9.1.2.3 Safety Evaluation

Design of this storage facility in accordance with NRC Regulatory Guide 1.13, Revision 1, December 1975, ensures a safe condition under normal and postulated accident conditions. The spent fuel storage racks are designed to prevent criticality by the means of physical separation of the fuel assemblies, administrative control of storage configurations in the racks (based on initial enrichment and minimum burnup requirements), special neutron absorbing material, and crediting of soluble boron (for both normal and accident conditions). Placement of fuel in the spent fuel pool outside of a rack module, such as in the gap between the rack module and the pool wall, is administratively controlled. Consideration of criticality safety analysis is discussed in Section 3.1.6.3 and Section 4.3.2.6.

The Spent Fuel Pool Cooling and Cleanup System is discussed in Subsection 9.1.3.

All surfaces that come into contact with fuel assemblies are made of materials that are resistant to corrosion during normal and emergency water quality conditions.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The Spent Fuel Pool Cooling and Cleanup System, a common system for both units, is designed in compliance with Title 10, Code of Federal Regulations, Part 50 Appendix A, General Design Criteria (GDC) 1, 2, 3, 4, 5, 44, 45, 46, 56, 61 and 63 [1], [2], [3], [4], [5], [6], [7] to perform the following principal functions:

1. To remove heat generated by stored spent fuel elements from the station's spent fuel pools

2. To maintain the clarity and purity of water in the spent fuel pools, the transfer canal, the wet cask pit, the RWST, and the refueling cavities

The calculations for the amount of thermal energy to be removed by the spent fuel pool cooling system are in accordance with BTP ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling" (Rev. 2).

Two cooling loops are provided, each capable of simultaneously servicing both of the station spent fuel pools. Two cleanup loops are also provided [14]. System design parameters are presented in Table 9.1-1.

The water depth above the top of the fuel assemblies as well as the removal of fission products and other contaminants by the system's purification loop limits the dose rate at the surface of the pools to 2.5 mr/hr.

9.1.3.1.1 Spent Fuel Pool Cooling

The Spent Fuel Pool Cooling and Cleanup System is designed to limit the temperature of the spent fuel pools in the following cases:

1. Maximum Design Condition

   The maximum design condition bounds the maximum normal heat loads which occur during refueling outages (RFOs). Temperature limits are in accordance with the ACI Code and ANSI N210.

   The spent fuel pool bulk water temperatures are maintained at less than 150^{o}F for normal operation based on decay heat generation from a normal full core offload at 125 hours after shutdown, plus decay heat from the opposite unit's last refueling discharge plus decay heat from fuel assemblies from a maximum number of previous refuelings in both pools. At least 193 spaces in the spent fuel pools are assumed to remain available to accept one full core in accordance with ANSI N18.2 [15].

   In actual practice, the 193 spaces assumed to remain available for one full core offload may be used for fuel assembly storage, with the exception of 32 fuel storage locations which must remain available for dry cask offloading in accordance with the requirements of the Cask Technical specifications. Based on the conservative assumptions in the

design conditions and the insignificant decay heat from additional assemblies in these in these spaces, the analyses remain valid even when these spaces are used for assembly storage.

Back to back refueling outages are assumed for conservatism. The first outage is assumed to be complete prior to the start of the second. The start of the second outage is conservatively assumed to begin 4 months after the first. A normal full core offload is conservatively assumed to start at 75 hours after the reactor is subcritical and complete at 125 hours. Off-load rates do not affect the analyses. Refueling discharges (which remain in the pools after refueling) are assumed to be up to 96 fuel assemblies for 18 month fuel cycles. The outages are assumed to begin and end during normal refueling periods (September 15th through May).

The SSI conditions are assumed to be representative of one unit operation at full power and one unit shutdown during normal refueling periods.

The normal design SFP HX outlet temperature is 140^{o}F to protect the resins in the cleanup system.

2. Maximum Summer Design Conditions

The maximum summer design condition bounds the maximum normal heat loads which occur during normal power operation of both units. Temperature limits are in accordance with the ACI Code and ANSI N210.

The spent fuel pool water temperatures are maintained at less than 150°F for normal operation based on decay heat from the most recent refueling discharge at the end of the outage plus decay heat from the opposite unit's previous refueling discharge plus decay heat from a maximum number of refuelings in both pools. At least 193 spaces in the spent fuel pool are assumed to remain available to accept one full core in accordance with ANSI N18.2 [15]. In actual practice, the 193 spaces assumed to remain available for one full core offload may be used for fuel assembly storage, with the exception of 32 fuel storage locations which must remain available for dry cask offloading in accordance with the requirements of the Cask Technical Specifications. Based on the conservative assumptions in the design conditions and the insignificant decay heat from additional assemblies in these spaces, the analyses remain valid even when these spaces are used for assembly storage.

The SSI temperature is assumed to be normal maximum (102°F).

The normal design spent fuel pool heat exchanger outlet temperature is 140°F to protect the resins in the cleanup system.

3. Abnormal Maximum Design Conditions

The abnormal maximum design condition bounds the abnormal heat load from an emergency core offload (ECO) from either unit immediately after back to back refuelings of both units.

The spent fuel pool water temperatures are maintained at less than 212°F for two loop operation based on an emergency core offload 125 hours after shutdown, plus the most recent refueling discharge 31 days prior to shutdown, plus the opposite unit's previous refueling discharge 5 months prior to shutdown, plus decay heat from a maximum number of previous refuelings in both pools.

Two spent fuel cooling loops are assumed to be available if required to meet temperature limits for this case; a single active failure need not be considered for an emergency core offload. Also, no other coincident events are assumed.

The SSI temperature is assumed to be normal maximum (102°F).

For fuel assembly loading in the spent fuel pools versus time, see Table 9.1-4.

One train operation is not normal during maximum design conditions. For the maximum normal heat load with normal cooling systems in operation, and assuming a single active failure, the design maximum pool temperature is 200°F; however, the design spent fuel pool heat exchanger outlet temperature is 140°F to protect the resins in the cleanup system. The level in the pools is maintained by makeup from the Reactor Makeup Water System which also meets the single active failure criterion.

Spent Fuel Pool Cooling to one or both pools could be lost temporarily due to an upset, emergency or faulted plant condition. There is sufficient time to restore forced spent fuel pool cooling prior to boiling. The spent fuel pool cooling system is designed to maintain water temperatures less than 212°F for one loop operation during and after plant upset, emergency, and faulted conditions coincident with maximum design or maximum summer design conditions.

The spent fuel pool water temperature in the above cases is based on the corresponding component cooling water temperature at the inlet to the spent fuel pool heat exchanger. The maximum component cooling water supply temperature is 122°F during normal cooldown with Residual Heat Removal System operation. This condition coincident with maximum spent fuel pool heat loads is considered an unlikely event which is expected to result in a small temperature increase for a short period of time during the transient.

Table 9.1-4 provides the number of spent fuel assemblies in spent fuel pools by refueling outage. Table 9.1-1 provides the decay heat parameters for the three design conditions above corresponding to the spent fuel storage in Table 9.1-4.

9.1.3.1.2 Water Purification

Two purification loops are provided for removal of fission products and other contaminates by means of filtration and ion exchange. Each purification loop is capable of purifying flow from either the spent fuel pool cooling pumps or the refueling water purification pumps. The use of two loops ensures maintenance of acceptable activity and purity levels in the spent fuel pools in the event of failure of one loop. Each purification loop limits the activity of fission and corrosion products in the spent fuel water to a maximum of 5×10^{-9} Ci/cm^3, exclusive of tritium. Purification is sufficient to permit unrestricted access to the spent fuel storage area.

The optical clarity of the spent fuel pool water surface is maintained by use of the skimmer, strainer, and skimmer filter of the system. The purification loops similarly provide for cleanup of water in the Refueling Water Storage Tank, refueling cavities, transfer canal, and the cask loading pits.

Evaporation and gaseous activity released to the atmosphere from the spent fuel pools are controlled by an air sweep system which provides a high-velocity air curtain across the pools (see Section 9.4.2).

9.1.3.2 System Description

The Spent Fuel Pool Cooling and Cleanup System consists of two cooling loops, two purification loops, and one surface skimmer loop. The system flow diagram is shown on Figure 9.1-13. Each cooling loop includes a pump, heat exchanger, and associated piping, valving, and instrumentation. Heat is transferred via the spent fuel pool heat exchanger to the Component Cooling Water System (CCWS). Depending on heat loads the system can be lined up with each train of equipment cooling one or both pools. If a heat exchanger is out of service (ie. CCW flow isolated) the affected train can be used for mixing by lining up to both pools. When heat loads are low a single train can be used to cool both pools.

Except during abnormal design conditions, one spent fuel pool cooling water pump and one spent fuel heat exchanger are capable of cooling both pools in the event that one train is out of service. Spent fuel pool cooling pump 01 (Train A) is protected from low suction pressure by the Train A level instruments in Pool 1. Spent fuel pool cooling pump 02 (Train B) is protected from low suction pressure by the Train B level instruments in Pool 2. Both spent fuel cooling pumps 01 and 02 (Train A and Train B) are protected with identical devices. These devices have 2 out of 2 logic which protects the pumps from erroneous trips. Redundant instruments require an instrument failure or LO-LO level pump trip signal from each device to trip the pumps. Pump 1 protection is provided by level instrumentation located in pool 1, therefore it is preferred that an operating pump be aligned to its associated pool (pump 1 to pool 1 and pump 2 to pool 2). When the pools are cross connected the pool levels will remain approximately equal and the pump protection provided by the LO-LO pool level pump trip will protect the cross connected pump. The suction lines, protected by spent fuel pool suction screens, are located approximately four ft below the normal spent fuel pool water level. The return lines terminate approximately six ft above the fuel assemblies, which prevents siphoning below this point in the event of a pipe break. To further ensure that siphoning does not occur, each return line contains an antisiphon hole approximately six in. below the low water level which corresponds to 12 in. below the normal water level.

Wide range spent fuel pool instrumentation is installed for monitoring SFP level in compliance with Order EA-12-051. The SFP level instrumentation is used as part of the mitigation strategies implemented for NRC Order EA-12-049. See Sections 1.2.5.1 and 1.2.5.2 for additional information.

During heat removal operations, a portion of the spent fuel pool water may be diverted through a demineralizer and filter in either of the purification loops to maintain spent fuel pool water clarity and purity. Transfer canal water may also be circulated through a purification loop by opening either of the two spent fuel pool gates and opening the valves in the cooling loop discharge lines to the transfer canal. In addition, the water in the transfer canal or the cask pits may be purified

by aligning the cask pit and transfer canal drain pump to take suction from the pit or canal and to discharge through a purification loop and back to the same pit or canal.

To allow maintenance of the fuel transfer equipment, the transfer canal is drained by the cask pit and transfer canal drain pump. The transfer canal water is pumped through the purification loop and discharged into the recycle holdup tank, which is part of the Boron Recycle System (BRS). After maintenance, an auxiliary discharge is provided in the BRS to return water to the refueling transfer canal using the recycle evaporator feed pumps.

The cask pits are drained in a similar manner. The cask pit and transfer canal drain pump impels the water through the purification loop and into the recycle holdup tank and returns water to the pits by way of the recycle evaporator feed pumps.

The demineralizer and filter are isolated manually from the heat removal portion of the system. The purification equipment can thus be used to maintain refueling water purity while spent fuel pool heat removal operations proceed simultaneously. Connections are provided so that the refueling water may be pumped from either the Refueling Water Storage Tanks or the refueling cavities through a filter and demineralizer and discharged back to either the refueling cavities or the Refueling Water Storage Tanks. Purification flow is obtained by way of the refueling water purification pump.

The valve arrangement of the purification loops is such that either loop may be used to maintain refueling water purity while the heat removal portion of the system is isolated manually. It is also possible to simultaneously use one purification loop for spent fuel pools and one purification loop for refueling water.

To further assist in maintaining water clarity in the spent fuel pools, the water surface is cleaned by a skimmer loop. Water is removed from the surface by the skimmers, pumped through a strainer and filter, and then returned to the pool at remote locations from the skimmers. Temporary equipment is used as required to assist in refueling cavity cleanup. The refueling cavity skimmer system is no longer used.

The spent fuel pools are filled with water of approximately the same (or greater) boron concentration than the RWSTs. The RWST Technical Specification requirements are verified prior to entering the applicable MODE of operation should intermixing of the borated water occur between the RWST and the SFP during outages.

The reactor makeup system shown in Figure 9.2-5, as described in Section 9.2.3, is a seismic Category I system. Ventilation requirements are discussed in Section 9.4.2. Makeup water to compensate for evaporation losses is also available from the demineralized water supply or reactor makeup water system (see Section 9.2.3).

A secondary function of the spent fuel pool cooling and cleanup system is to form a small portion of a BR cross tie flowpath between the BRS and the SI systems. This Y-shaped flow path allows the efficient transfer of RHUT water to either RWST. See Figure 9.1-13 and Section 9.3.4.2 for a more detailed description of this cross tie.

9.1.3.2.1 Component Description

Codes and safety classifications for Spent Fuel Pool Cooling and Cleanup System components are given in Table 9.1-2.

All process lines shown on Figure 9.1-13 and identified as nuclear safety class are classified Seismic Category I. The boundary between the Seismic Category I piping and non-Seismic Category I piping coincides with the boundary between safety class piping and non-safety class piping. This separation appears on Figure 9.1-13 as safety class 3 to piping class 5 (NSS) transition except on vent, drain and test lines which are NNS downstream of the root valve.

All piping in contact with spent fuel pool water is made of stainless steel. The piping is welded except where flanged connections are used to facilitate maintenance.

For instrumentation applications, see Subsection 9.1.3.5.

9.1.3.3 Safety Evaluation

Spent fuel pool water is cooled by two redundant cooling loops, each of which contains a pump, heat exchanger, piping, valves, and instrumentation. In the event of a failure of spent fuel pool cooling pump or heat exchanger, the other loop ensures the continuity of effective cooling.

In case of spent fuel stored in both pools or a closely spaced refueling of both reactors, the two cooling loops may be used. In the event of a failure of one loop, the second loop ensures a minimum cooling and limits the water temperature to the cleanup system to less than 140°F to protect the demineralizer resins.

To detect leakage through the spent fuel pool liner welds, a channel is provided in back of the welds to form a leak chase. Concrete troughs are formed under the welds in the floor plate. Sections of welds which are leaking can be determined by observing which leak chase the water is coming from before the leak chases merge into a common drain header. Any leaks detected will be evaluated.

Furthermore, as indicated in Table 9.1-1, the pool capabilities are sufficiently large so that an extended cooling outage is required before pool temperatures reach 212°F. Thus the system can be shut down safely for reasonable time periods for maintenance or replacement of malfunctioning components. The effect of the evaporation rate from the pools on humidity are described in Section 9.4.2.

The suction lines inside the spent fuel pools are positioned to take suction four ft below the normal water level in order to minimize vortexing and the possibility of floating debris entering the system. Return lines from the spent fuel pool heat exchangers are located so that cooled water is discharged downward approximately six ft above the fuel assemblies. This ensures adequate dispersion of the cooled water around the stored spent fuel assemblies. The suction and return lines are located on opposite sides of the pools to prevent channeling and to obtain maximum circulation.

To protect against loss of water from the spent fuel pools, the spent fuel pool cooling pump suction lines penetrate the pool wall and terminate approximately four ft below the normal water level and the return lines terminate six ft above the fuel assemblies. The return lines contain

antisiphon holes. This arrangement precludes gravity draining of the pools in the event of a pipe break and ensures that sufficient shielding is maintained.

There are no drain lines connected to the pool. Appropriate redundancy, including a seismic Category I source, is provided for makeup water to the pools. Draining of either pool below the design water level is not considered credible. The rate of makeup water is greater than the rate of water loss. The radiological evaluation of the cleanup system is presented in Chapters 11 and 12.

9.1.3.4 Inspection and Testing Requirements

The active components of this system are in either continuous or intermittent use during normal plant operation. Periodic visual inspections and preventive maintenance are conducted as necessary. All components are accessible for periodic inspection except one section of each cooling pump suction line and one section of the cooling water return line. These sections, of all-welded construction, are embedded in concrete in the vicinity of the spent fuel pool and cask storage area.

To ensure that proper operational conditions exist for the spent fuel pool, periodic chemical analyses and operational surveillance shall be performed when this system is in use. Chemical analyses will be performed weekly for determining concentrations of chloride, fluoride and boron. Radioactivity levels and pH will be determined, as a minimum, on a weekly frequency. The chemical limits used in the monitoring of the spent fuel pool are, as follows:

Chlorides	0.15 ppm (maximum)
Fluorides	0.15 ppm (maximum)
PH	Variable
Boron Concentration	2400 ppm (minimum)
Radioactivity Levels	Activity levels shall be maintained as low as reasonably achievable (ALARA)

The bases for these limits are to minimize the potential for corrosion attack, to ensure the proper reactivity control and to maintain the radioactivity levels as low as reasonably achievable (ALARA).

For information on the sampling and monitoring of the spent fuel pool demineralizers and filters, see Section 12.2.1.2.2.

9.1.3.5 Instrument Requirements

The instrumentation for the Spent Fuel Pool Cooling and Cleanup System is discussed in the following paragraphs.

1. Temperature

Local temperature indicators are provided at the spent fuel panel for spent fuel pools and the refueling cavities and also provided at the outlet of the spent fuel pool heat exchanger.

Annunciation is given in the spent fuel pool panel and a common trouble alarm in the control room when the normal temperature is exceeded.

2. Pressure

Pressure gauges are provided at the discharge of the pumps used in this system. Local differential pressure indicators are connected across the spent fuel pool filter, spent fuel pool skimmer filter, spent fuel pool demineralizer and resin trap, with an alarm on the spent fuel pool panel and a common trouble alarm in the control room.

3. Flow

Local indicators are provided to indicate the flows through the purification loop and in the spent fuel pool return lines. Low flow in the pool return lines is also alarmed in the spent fuel pool panel and a common trouble alarm in the control room.

4. Level

The spent fuel pool water levels are measured to give alarms in the local panel and a common alarm in the Control Room for high and low water level. A LO-LO water level trip is provided for cooling pump protection. The level instrumentation is designed to provide 2 out of 2 logic for cooling pump protection on LO-LO water level. Continuous level indication is provided to the Control Room via the plant computer.

See FSAR Section 1.2.5.2 for a description of the wide range level instrumentation for beyond design basis events.

5. Radiation

Area radiation monitors are located in the fuel pool area. See Section 12.3 for a description of the area radiation monitors.

9.1.4 FUEL HANDLING SYSTEM

9.1.4.1 Design Bases

The fuel handling system (FHS) consists of equipment and structures used for conducting the refueling operation in a safe manner; this system conforms to GDC 61 and 62 of 10 CFR Part 50, Appendix A.

The following design bases apply to the FHS:

1. Fuel handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operation.

2. Handling equipment has provisions to avoid dropping of fuel handling devices during the fuel transfer operation.

3. Handling equipment used to raise and lower spent fuel has a limited maximum lift height so that the minimum required depth of water shielding is maintained.

4. The Fuel Transfer System (FTS), where it penetrates the Containment, has provisions to preserve the integrity of the Containment pressure boundary.

5. Criticality during fuel handling operation is prevented by the geometrically safe configuration of the fuel handling equipment.

6. In the event of an SSE, handling equipment cannot fail in such a manner as to damage seismic Category I equipment.

7. The inertial loads imparted to the fuel assemblies or core components during handling operations are less than potential damage-causing loads.

8. Physical safety features are provided for personnel who operate handling equipment.

9.1.4.2 System Description

The FHS consists of the equipment needed for the refueling operation on the reactor core. Basically, this equipment is comprised of fuel assemblies, core component and reactor component hoisting equipment, handling equipment, and a FTS. The structures associated with the fuel handling equipment are the refueling cavity, including the fuel storage area in the Containment, and the refueling canal, cask pit, and spent fuel pools in the Fuel Building.

In addition, heavy loads associated with fuel handling are lifted by the Containment Polar Crane and by the Fuel Building Overhead Crane. The Containment Polar Crane is used to remove the reactor vessel head and internals for refueling. The Fuel Building Overhead Crane is used to lift new and spent fuel casks.

The elevation and arrangement drawings of the fuel handling facilities are on Figures 1.2-12, 1.2-13, 1.2-18, 1.2-19, 1.2-38, 1.2-39, and 1.2-40.

9.1.4.2.1 Fuel Handling Description

New fuel assemblies received for initial refueling are removed one at a time from the shipping container and moved to the new fuel assembly inspection area. After completion of inspection, the acceptable new fuel assemblies are stored in the new fuel storage racks or are lowered by the new fuel elevator and stored in the spent fuel racks in the spent fuel pools.

The fuel handling equipment is designed to handle the spent fuel assemblies underwater from the time they leave the reactor vessel until they are placed in a container for shipment from the site. Underwater transfer of spent fuel assemblies provides an effective, economic, and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. The boric acid concentration in the water is sufficient to preclude criticality.

The refueling cavity in Containment is divided by a refueling gate into two areas: the reactor vessel area and Containment fuel transfer area. These areas are flooded only during plant shutdown for refueling. The entire refueling cavity is kept full of water whenever spent fuel is being moved or stored in it. The refueling gate is used to isolate either area of the refueling cavity for drain down if required. The associated fuel handling structures in the Fuel Building are divided by refueling gates into three areas: the fuel transfer canal, the cask pit, and the spent fuel pools. These areas are kept full of water whenever spent fuel is being moved or stored in them. The Containment fuel transfer area and the fuel transfer canal are connected by a fuel transfer tube which is fitted with a blind flange on the Containment end and a gate valve on the Fuel Building end. The blind flange is in place except during refueling to ensure Containment integrity. Fuel is carried through the tube on an underwater transfer car.

Fuel is moved between the reactor vessel and the Containment fuel transfer area by the refueling machine. A control rod cluster changing fixture is provided for transferring control elements from one fuel assembly to another. Control elements may also be changed in the Fuel Building. The FTS is used to move fuel assemblies between the Containment Building and the Fuel Building.

In the Fuel Building, fuel assemblies are moved about by the fuel handling bridge crane and the Fuel Building overhead crane. Fuel movements are limited such that no more than two fuel assemblies are outside of an approved shipping container, fuel inspection station, storage rack or fuel transfer tube at any one time and a minimum 12-inch edge-to-edge distance is maintained between such assemblies. For underwater fuel handling operations, a minimum 12-inch edge-to-edge distance is maintained between any assembly that is outside of an approved shipping container, storage rack, or fuel transfer tube and any other assembly; including those in shipping containers, storage racks, or fuel transfer tubes. These spacing requirements, in combination with fuel assembly design and enrichment characteristics, assure that Keff remains below 0.95 for dry (unmoderated) as well as flooded (unborated water) conditions.

A discussion of the spent fuel pool Region I and Region II oversized inspection cells is contained in the Technical Specification Bases 3.7.17.

Decay heat, generated by the spent fuel assemblies in the spent fuel pools, is removed by the Spent Fuel Pool Cooling and Cleanup System. After a sufficient decay period, the spent fuel assemblies are removed from the fuel racks and loaded into a spent fuel shipping cask for removal from the site.

9.1.4.2.2 Refueling Procedure

The refueling operation follows a detailed procedure which provides safe and efficient refueling. Prior to initiating refueling operation, the RCS is borated and cooled down to refueling shutdown conditions as specified in the Technical Specifications. Criticality protection for refueling operations is specified in the Technical Specifications. The following significant points are ensured by the refueling procedure:

1. The refueling water and the reactor coolant contain sufficient boron so that with the negative reactivity of control rods, the core can be maintained shutdown during the refueling operation. It is also sufficient to maintain the core subcritical in the unlikely event, that all of the rod cluster control assemblies (RCCA) were removed from the core.

2. The water level in the refueling cavity is high enough to keep the radiation levels within acceptable limits when the fuel assemblies are being removed from the core.

The refueling operation is divided into four major phases: preparation, reactor disassembly, fuel handling, and reactor assembly. A general description of a typical refueling operation through the phases is given as follows:

1. Phase I Preparation

   The reactor is shut down and borated to Refueling shutdown condition as specified in the Technical Specifications. The coolant level in the reactor vessel is lowered to a point slightly below the vessel flange and the fuel transfer equipment and refueling machine are checked for proper operation.

2. Phase II Reactor Disassembly

   The external air duct is removed from the CRDM Ventilation System's Air Handling Units and plenum (Unit 1 only). Cables are disconnected to allow removal of the roll-away missile shield (Unit 2 only) and the vessel head. Air ducts (Unit 2 only) and insulation are removed from the vessel head. The refueling cavity is then prepared for flooding by checking the underwater lights, tools, and FTS, closing the refueling canal drain lines, and removing the blind flange from the fuel transfer tube. With the refueling cavity prepared for flooding, selected portions may be filled in preparation for vessel head removal. With the water level providing a safe shielding depth (see Subsection 9.1.4.3.4), the vessel head is taken to its storage pedestal. The remainder of the refueling cavity is flooded to the depth specified in the Technical Specifications. The control rod drive shafts are disconnected and, with the upper internals, are removed from the vessel. The fuel assemblies and RCCA are now free from obstructions and the core is ready for refueling.

3. Phase III Fuel Handling

   The refueling sequence is started with the refueling machine. The positions of partially spent assemblies are changed, and new assemblies are added to the core.

   The refueling machine is positioned over a selected fuel assembly and the fuel assembly is lifted to a predetermined height sufficient to clear the reactor vessel. The refueling machine is moved to align the fuel assembly with the fuel transfer system and the fuel assembly is placed into the fuel assembly container of the transfer car. The container is pivoted to the horizontal position by the lifting arm and is moved through the fuel transfer tube to the Fuel Building by the transfer car. The fuel assembly container is pivoted to the vertical position. The fuel assembly is unloaded by the fuel handling bridge crane using the spent fuel handling tool and placed in its designated storage location.

   Fuel and insert components can be shuffled in either Containment or Fuel Building using the tools designated for components.

   Any partially spent fuel not designated for removal from the core is shuffled to its desirable new location for the next cycle as the reactor is reloaded. Fuel is reloaded into the reactor by reversing the steps necessary for fuel removal.

4. Phase IV Reactor Assembly

Reactor assembly, following refueling, is essentially achieved by reversing the operations given in Phase II, Reactor Disassembly.

9.1.4.2.3 Component Description

1. Refueling Machine

The refueling machine (Figure 9.1-3) is a rectilinear bridge and trolley system with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and runs on rails set into the edge of the refueling 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 down out of the mast to grip the fuel assembly and the tube is long enough so that the upper end is still contained in the mast when the gripper end contacts the fuel. Mounted on the trolley, a winch 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 refueling machine uses a programmable logic controller and a variable frequency control system that gives stepless variable speeds from zero to full speed.

All controls for the refueling machine are mounted on a console in the trolley. The bridge and trolley are positioned by use of indexing plates. A video camera and television monitor provides a picture for the operator of the north-south indexing plate. The drives for the bridge, trolley, and winch are variable speeds and include a separate inching control on the winch. The maximum speeds for the bridge, trolley and main hoist are 60, 26 and 26 ft/min respectively. The auxiliary monorail hoist on the refueling machine has a two-step magnetic controller to give hoisting speeds of approximately 7 and 20 ft/min.

Electrical interlocks and limit switches on the bridge and trolley drives prevent damage to the fuel assemblies. The winch is also provided with limit switches to prevent a fuel assembly from being raised above a safe shielding depth should the limit switch fail. In an emergency, the bride, trolley, and winch can be operated manually using a handwheel on the motor shaft.

The refueling machine is provided with a television system which permits viewing of fuel assembly positions and fuel movements.

Additionally, the refueling machine is provided with a fuel sipping system. This “In-Mast Sipping System” is a set of hardware that provides a means of performing on-line, quantitative leak testing of fuel assemblies in the refueling mast during normal fuel handling operations. Included in the sipping system are: mechanical equipment connected to the refueling machine mast, a control module, a pump and valve module, and a detection and recording module.

2. Fuel Handling Bridge Crane

The Fuel Handling Bridge Crane (FHBC) (Figure 9.1-4), is a wheel-mounted walkway spanning the spent fuel pools, which carries two electric trolley hoist on an overhead structure. This crane is used for handling fuel assemblies within the spent fuel pools, refueling canal, and cask pits by means of a long-handled tool suspended from the hoist. A load monitoring device is attached to the crane between the hoist hook and the fuel handling tool for monitoring fuel assembly loads. The hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

Although the FHBC has two trolley-hoist assemblies, redundant and diverse interlocks are provided to ensure the FHBC will not have the ability to suspend more than one load (e.g. fuel assembly) at a time. No bypass functions are provided for these interlocks.

The fuel handling bridge and trolley have single speed motor controlled by a variable frequency controller. The programmed maximum speed of the bridge and trolley is 40 feet per minute. The hoist has a single speed motor controlled by a variable frequency controller. The hoist speeds are limited to 3 feet per minute in the slow zones and 21 feet per minute outside slow zones.

For loads equal to or greater than the combined weight of a fuel assembly and the handling tool, the maximum hoist speed is automatically limited to 21 fpm. For loads less than the combined weight of a fuel assembly and the handling tool, the maximum hoist speed is automatically allowed to operate at the higher ultra speed of 31.5 fpm.

To ensure safe handling of the fuel, the FHBC allows bridge movement only if the hoist is at zero speed. The bridge can only operate at microspeed (3 feet per minute) if the hoist is not at full up position when loaded with a Fuel Assembly and the Fuel Assembly Handling Tool. Hoist operation is not permitted while either the bridge or the trolley is moving. Trolley operation is not permitted while the hoist is moving. The trolley can only operate at microspeed if the hoist is not at the full up position when loaded with a Fuel Assembly and the Fuel Assembly Handling Tool. Bridge and trolley motion can only be combined if the hoist is completely up or if the suspended weight is less than the combined weight of a Fuel Assembly and the Fuel Assembly Handling Tool (i.e. equal to 1200 pounds or less). If this is not the case, only the first activated movement (bridge or trolley) is active.

Additional automatic features include:

During fuel movement, when the weight of a fuel assembly is detected, bridge and trolley motion is not possible while hoisting or lowing unless the fuel assembly is indicated to be clear of the rack or upender.

The fuel transfer cart upender cannot be raised unless these functions are satisfied:

Bridge axis is clear of the upender.

Trolley axis is clear of the upender.

Hoist with fuel is fully raised.

Hoist is above the lower slow zone without fuel.

Trolley movement is limited in travel at each end of the girder. The trolley automatically slows then stops as the trolley approaches the end stops. This is an added feature which prevents the trolley from impacting the end stops at full speed.

Bridge speed is limited in travel at each end of the runway. The bridge automatically slows then stops the bridge movement as it approaches the end stops.

The FHBC control station is automatically moved by a motor assembly so that it remains with the hoist-trolley it is currently controlling in order to assure that the fuel handler remains with the fuel assembly being moved with the crane. A switch to allow the fuel handler to override this feature is provided.

In addition to these automatic features, all fuel handling operations are carried out using strict administrative controls.

3. Containment Fuel Handling Bridge Crane

The Containment fuel handling bridge crane is a wheel-mounted walkway spanning the Containment upender area, which carries an electric trolley hoist on an overhead structure. The Containment fuel handling bridge crane is used exclusively for handling fuel assemblies within the Containment 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.

4. New Fuel Elevator

The new fuel elevator (Figure 9.1-5) consists of a box-shaped elevator assembly with its top end open and sized to house one fuel assembly. It is used to lower a new fuel assembly to the bottom of the cask pit refueling equipment area where it is transported to the storage racks by the fuel handling bridge crane.

It is also used to raise the fuel assembly for manipulation during fuel reconstitution. As a precaution, additional restraints are established during this mode to prevent the elevator from raising the fuel within 10 feet of fuel pool water level.

5. Fuel Transfer System

The FTS (Figure 9.1-6) includes an underwater, electric motor driven transfer car that runs on tracks extending from the refueling canal, through the transfer tube, and into the Containment fuel storage area; a hydraulically actuated lifting arm is at each end of the transfer tube. The fuel container in the Containment fuel storage area receives a fuel assembly in the vertical position from the refueling machine. The fuel assembly is then lowered to a horizontal position for passage through the transfer tube. After passing through the tube, the fuel assembly is raised to a vertical position for removal by a tool suspended from a hoist mounted on a fuel handling bridge crane in the Fuel Building.

The fuel handling bridge crane then moves to a storage loading position and places the spent fuel assembly in the spent fuel storage racks.

During reactor operation, the transfer car is stored in the fuel storage area. A blind flange is bolted on the Containment end of the transfer tube to seal the reactor containment. The terminus of the tube outside the Containment is closed by a gate valve.

6. Rod Cluster Control Changing Fixture

The rod cluster control changing fixture is supplied for periodic rod cluster control element inspections and for transfer of rod cluster control elements from one fuel assembly to another if this is ever required (Figure 9.1-7). The major subassemblies which comprise the changing fixture are the frame and track structure, the carriage, the guide tube, the gripper, and the drive mechanism. The carriage is a movable container supported by the frame and track structure. The tracks provide a guide for the four flanged carriage wheels and allows horizontal movement of the carriage during changing operation.

In order to locate each of the three carriage compartments directly below the guide tube, the positioning stops are provided on both the carriage and frame. Two of these compartments are designed to hold individual fuel assemblies; the third is made to support a single rod cluster control element. Situated above the carriage and mounted on the refueling canal wall is the guide tube. It provides for the guidance and proper orientation of the gripper and rod cluster control element as they are being raised and lowered. The gripper is a pneumatically actuated mechanism responsible for engaging the rod cluster control element. It has two flexure fingers which can be inserted into the top of the rod cluster control element when air pressure is applied to the gripper piston. Normally the fingers are locked in a radially extended position. Mounted on the operating deck is the drive mechanism assembly, which is made up of the manual carriage drive mechanism, the revolving stop operating handle, the pneumatic selector valve for actuating the gripper piston, and the electric hoist for elevation control of the gripper.

7. Spent Fuel Assembly Handling Tool

The spent fuel assembly handling tool (Figure 9.1-8) is used to handle new and spent fuel assemblies in the fuel storage areas. It is a manually actuated tool, suspended from the fuel handling bridge cranes, which uses four cam actuated latching fingers to grip the underside of the fuel assembly top nozzle. The operating handle to actuate the fingers is located at the top of the tool. When the fingers are latched, a pin is inserted into the operating handle which prevents the fingers from being accidently unlatched during fuel handling operations.

8. New Fuel Assembly Handling Tool

The new fuel assembly handling tool (Figure 9.1-9) is used to lift and transfer fuel assemblies from the new fuel shipping containers to the new fuel storage racks or new fuel elevator. A manually actuated tool suspended from the Fuel Building overhead crane or fuel handling bridge crane uses four cam-actuated latching fingers to grip the underside of the fuel assembly top nozzle. The operating handles which actuate the fingers are located on the side of the tool. When the fingers are latched, the safety screw is turned in to prevent the accidental unlatching of the fingers.

9. Reactor Vessel Head Lifting Device

The reactor vessel head lifting device consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations. Attached to the head lifting device are the monorail and hoists for the reactor vessel stud tensioners.

10. Reactor Internals Lifting Device

The reactor internals lifting device (Figure 9.1-10) is a structural frame suspended from the polar crane. The frame is lowered onto the guide tube support plate of the internals and is mechanically connected to the support plate by three breech-lock-type connectors. Bushings on the frame engage guide studs in the vessel flange to provide guidance during removal and replacement of the internals package.

11. Reactor Vessel Stud Tensioner

The stud tensioners (Figure 9.1-11) are used to secure the head closure joint at every refueling. The stud tensioner is a hydraulically operated device that uses oil as the working fluid. The device permits preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners minimize the time required for the tensioning or unloading operation. The studs are tensioned to their operational load in an approved sequence to prevent high stresses in the flange region and unequal loadings in the studs. Relief valves on each tensioner prevent overtensioning of the studs because of excessive pressure.

12. Containment Fuel Storage Rack

This rack is located in the Containment fuel transfer area to facilitate fuel assembly movements. This rack may be used for interim storage of new or spent fuel. Spent fuel or irradiated components (i.e., RCCAs, BPRAs, Thimble Plugs, Secondary Sources, and Reactor Vessel Irradiated Specimens) are not stored in this rack unless the entire refueling cavity is full.

13. Not used.

14. Containment Polar Crane

The Containment Polar Crane is a single trolley traveling bridge crane rotating on a single rail circular track and provided with cab and radio control stations. See Figure 9.1-14.

The crane is used for construction lifts and for refueling and maintenance operations with occasional lifts at the maximum rated crane lifting capabilities. See Section 9.1.4.3.2 for the Polar Crane seismic considerations. See Section 3.8.3.4.10 for the applicable industry standard.

A telescopic jib crane is also provided inside Containment for refueling and maintenance. It is supported by a structural steel support mounted on east-west divider wall between S.G. compartments at elevation 905'-9". Both the Polar Crane and Telescopic Jib Crane operating at the same time constitutes Multi-crane Operation which requires special

administrative controls to reduce the likelihood of unplanned crane interactions. See Section 3.8.3.1.2 for a description of the administrative controls and design features for the Polar Crane and the Telescopic Jib Crane.

During normal operation of the plant the crane is set in the pre-established parked position (see Figure 1.2-8). Prior to entering outages, crane operation is required to perform certain inspections, functional checks and preventative maintenance (PM) activities. The polar crane may be energized during MODES 1-4. The crane will remain essentially in the park position (FSAR Figure 1.2-8) during MODES 1 through 4 unless maintenance is being performed on it. The Containment Access Rotating Platform is allowed to be used for maintenance on (including modifications to) the Polar Crane in Modes 1 through 4. During Polar Crane maintenance, the polar crane may be operated through its full 360 degree travel range. The trolley is to remain as close to the liner plate/ crane rail as possible. The hook is not to be moved outside the park position. The hook should not be placed in a location to impact any safety related SSC. No load will be carried by the polar crane during MODES 1-4. A compensatory measure (a dedicated person in direct communication with the control room, at the power supply disconnect outside containment) is established to prevent inadvertent operation of crane due to High Energy Line Break in containment.

The main components of the Containment Polar Crane are:

a. Bridge with welded box-section-type girders, cab footwalk and trolley rails (see Figure 9.1-14, Sheet 1 of 3).

b. Bridge end sills and trucks.

c. Crane trolley consisting of:

   Frame

   Mainhoist machinery (175 tons/475 tons)

   Auxiliary hoisting machinery (20 tons)

   Upper sheave assembly

   Bottom block

   Upper equalizer bar

   Main and auxiliary drums

   Main and auxiliary ropes

   (see Figure 9.1-14, Sheet 2 of 3)

d. Electrical equipment

e. Control system equipment

15. Fuel Building Overhead Crane

The Fuel Building Overhead Crane is the primary means of transporting nuclear fuel in and out of the fuel handling area of the Fuel Building. Its range includes the spent fuel cask loading area, the new fuel storage pit, the cask handling area, the new fuel receiving area and the railroad loading and unloading area. (see Figures 1.2-9 and 1.2-40). The crane is designed as a traveling bridge crane with a single trolley and is provided with a cab.

Reference [17] provides information regarding the design of the Fuel Building Overhead Crane.

16. Refueling Gates

The refueling gates consist of lift type gates and hinged gates. Lift gates are used for prohibiting a transfer of water across the gate from one section of the refueling cavity or canal to the other sections of the cavity or canal. Hinged gates are used for prohibiting the exit of water from the spent fuel pool into an empty refueling canal. The gates are designed to withstand forces resulting from the simultaneous combination of seismic forces and the seismically induced hydrodynamic fluid pressure and maintain their leakproof integrity.

The refueling gates are provided with inflatible seals to maintain a watertight barrier under all conditions. The seals are designed to be passive devices and do not require a continuous pneumatic supply. A minimum of one seal suitably isolated from its NNS pneumatic supply is required to perform a gate's Nuclear Safety functions. See Section 9.3.1 for a description of the Compressed Air Systems.

During spent fuel transfer in the transfer canal adjacent to a pool, the spent fuel pool involved in the transfer is normally open and full to its normal operating level. If an adjacent pool (not involved in the transfer) is below its normal operating level and is isolated by its gate from the canal, the level of water in the pool is maintained such that any failure of the gate would not result in loss of spent fuel cooling or in excessive radiation levels from a fuel assembly in transit.

17. Rod Cluster Control Change Tool

The RCC change tool is portable and functions in a manner similar to the stationary RCC change fixture in removing or replacing an RCC in a fuel assembly.

The RCC change tool consists of three basic assemblies. The first is the guide tube. The guide tube is the same design as the new RCC change fixture guide tube. Guide plates in the lower end of the tube provide guidance for the gripper and the RCC. The upper portion of the guide tube has guides that orient and align the gripper assembly as the RCC is removed from the fuel assembly.

Second, above the guide tube is the support tube that gives the proper length to the tool and provides support for the gripper actuator and protection for the lift cable. The gripper actuator consists of a pneumatic system that provides for the engagement and disengagement of the latch mechanism to the RCC spider. The gripper mechanism is

similar to the present RCC change fixture. A limit switch will indicate the engage or disengage position for the gripper actuator as an additional safety feature.

Third, the drive mechanism consists of an electric-powered winch and an enclosure with controls for gripper actuation and winch operation. Limit switches prevent winch overtravel in either direction, and an overload switch protects the system during RCC lifting operations. Lamps on the enclosure face display the status of all switches.

The bottom of the tool is equipped with locating pins to orient the tool with respect to the fuel assembly nozzle.

18. Portable Underwater Lights

Portable underwater lighting adequate for safe handling of irradiated fuel assemblies is provided. These lights are used as needed in the refueling cavity, transfer canals, and spent fuel pools.

9.1.4.2.4 Applicable Design Codes

The design codes and standards used for the FHS are given in Subsection 9.1.4.3 and in Table 17A-1.

The Containment Polar Crane and the Fuel Building Overhead Crane are designed to seismic Category I requirements (as defined in Section 3.2 and listed in Table 17A-1) as discussed in Section 9.1.4.3.2.

As stated in Section 3.8.3.2.3, the criteria of CMAA Specification No. 70 are also used for the design, testing and maintenance of both cranes.

9.1.4.3 Safety Evaluation

9.1.4.3.1 Safe Handling

Design criteria for the FHS are as follows:

1. The primary design requirement of the equipment is reliability. A conservative design approach is used for all load bearing parts. Where possible, components are used that have a proven record of reliable service. Throughout the design, consideration is given to the fact that the equipment spends long idle periods stored in a high humidity and temperature atmosphere (up to 122°F and 100% humidity).

2. Except as otherwise specified, the refueling machine is designed and constructed in accordance with Crane Manufacturers Association of America, Inc. (CMAA) Specification 70.

   The fuel handling bridge crane designed and constructed in accordance with ASME NUM-1, Type B for Hoisting equipment and ASME NOG-1. Type II for Trolley and rest of the crane.

3. Design load for the refueling machine and Fuel Building fuel handling bridge crane shall be normal dead and live loads plus maximum hoist load.

4. The allowable stresses for the refueling machine structures supporting the weight of the fuel are as specified in Subarticle XVII-2200 of the ASME B&PV Code, Section III, Appendix XVII (1974).

   Hoisting components loaded in simple tension have an allowable stress of 0.20 ultimate stress maximum.

5. All components critical to the operation of the refueling equipment used over the reactor-vessel or located so that parts can fall into the reactor are assembled with the fasteners positively restrained from loosening under vibration. This is applicable to the following equipment: refueling machine, and the control rod drive shaft unlatching tool (critical fasteners).

   Spring- or tooth-type lockwashers are not considered as positive restraint. Fasteners on purchased components that cannot be wired or tackwelded are coated with an acceptable locking compound or, alternatively, are positively restrained by another acceptable method.

Industrial codes and standards used in the design of the fuel handling equipment are as follows:

1. The refueling machine conforms to applicable sections of CMAA Specification 70.

   The fuel handling bridge crane conforms to applicable sections of ASME NUM-1, Type B for Hoisting equipment and ASME NOG-1. Type II for Trolley and the rest of the crane.

2. The refueling machine's structural components conforms to the ASME B&PV Code, Section III, appendix XVII, Subarticle XVII-2200 (1974).

3. Electrical equipment conforms to the applicable standards and requirements of National Electrical Code, NFPA 70, and NEMA Standards MGI and ICS for design, installation, and manufacturing.

4. Materials conform to the specifications of ASTM standards.

5. Safety standards include the OSHA standards, 29 CFR Parts 1910 and 1926, including load testing requirements, the requirements and ANSI N18.2, NRC Regulatory Guide 1.29, and GDC 61 and 62.

The refueling machine design includes the following provision to ensure safe handling of fuel assemblies:

1. Electrical Interlocks

   a. Bridge, Trolley, and Hoist Drive Mutual Interlocks

Bridge, trolley, and winch drives are mutually interlocked, using redundant interlocks to prevent simultaneous operation of any two drives; therefore, they can withstand a single failure.

b. Bridge and Trolley Drive, Gripper Tube Up

Bridge and trolley drive operation is prevented except when the gripper tube up position switches are actuated. The interlock is redundant and can withstand a single failure.

c. Gripper Interlock

An interlock is supplied which prevents the opening of a solenoid valve in the air line to the gripper except when zero suspended weight is indicated by a force gauge. 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. This interlock is redundant and can withstand a single failure.

d. Excessive Suspended Weight

Two redundant excessive-suspended-weight switches open the hoist drive circuit in the up direction when the loading is in excess of 110 percent of a fuel assembly weight. The interlock is redundant and can withstand a single failure.

e. Hoist-Gripper Position Interlock

An interlock in the hoist drive circuit in the up direction permits the hoist to be operated only when either the open or closed indicating switch on the gripper is actuated. The hoist-gripper position interlock consists of two separate circuits that work in parallel so that one circuit must be closed for the hoist to operate. If one or both interlocking circuits fail in the closed position, an audible and visual alarm on the console is actuated. The interlock, therefore, is not redundant but can withstand a single failure since both an interlocking circuit and the monitoring circuit has to fail to cause a hazardous condition.

2. Bridge and Trolley Holddown Devices

Both refueling machine bridge and trolley are horizontally restrained on the rails by two pairs of guide rollers, one pair at each wheel location on one truck only. The rollers are attached to the bridge truck and contact the vertical faces on either side of the rail to prevent horizontal movement. Vertical restraint is accomplished by antirotation bars located at each of the four wheels for both the bridge and trolley. The antirotation bars are bolted to the trucks and, for the bridge restraints, extended under the rail flange; while, for the trolley restraints, they extend beneath the top flange of the bridge grider which supports the trolley rail. Both horizontal and vertical restraints are designed to adequately withstand the forces and overturning moments resulting from the SSE.

3. Design Load

The design load for structural components supporting the fuel assembly is the dead weight plus 4800 lb (three times the fuel assembly weight).

4. Main Hoist Braking System

The main hoists are equipped with two independent braking systems. A solenoid release-spring-set electric brake is mounted on the motor shaft. This brake operates in the normal manner to release, upon application of current to the motor, and set when current is interrupted. The second brake is a mechanically actuated load brake internal to the hoist gearbox that sets if the load starts to overhaul the hoist. It is necessary to apply torque from the motor to raise or lower the load. In raising, this motor cams the brake open; in lowering, the motor slips the brake allowing the load to lower. This brake actuates upon loss of torque from the motor for any reason and is not dependent on any electrical circuits. On the main hoist the motor brake is rated at 350-percent operating load and the mechanical brake at 300 percent.

5. Fuel Assembly Support System

The main hoist system is supplied with redundant paths of load support such that failure of any one components does not result in free fall of the fuel assembly. Two wire ropes are anchored to the winch drum and carried over independent sheaves to a load equalizing mechanism on the top of the gripper tube. In addition, supports for the sheaves and equalizing mechanism are backed up by passive restraints to pick up the load in the vent of failure of this primary support. Each cable system is designed to support 13,750 lb or 27,500 lb acting together.

The working load of fuel assembly plus gripper is approximately 2500 lb.

The gripper itself has four fingers that grip the fuel and support the fuel assembly weight.

The gripper mechanism contains a spring-actuated mechanical lock which prevents the gripper from opening unless the gripper is under a compressive load.

The following safety features are provided for in the FTS.

1. Transfer Car Permissive Switch

The transfer car controls are located in both Fuel and Reactor Building and the conditions of the lifting arm as well as transfer car are displayed on both consoles during normal operating condition of the consoles. The transfer car permissive interlock allows either operator to move the car in either direction as required. The operator however can only operate the lifting arm on side that the operator is located on.

Transfer car operation is possible only when both lifting arms are in the down position as indicated by the limit switches. The load system and frame up limit switch are the backups for the lifting arm interlock. Assuming the fuel container is in upright position in the Reactor Building, and the lifting arm interlock circuit fails in the permissive condition, neither operator can operate the car because of the PLC interlock and mechanical interlock. The interlock, therefore can withstand a single failure.

2. Lifting Arm - Transfer Car Position

   Two redundant interlocks allow lifting arm operation only when the transfer car is at the respective end of its travel and therefore can withstand a single failure.

   Of the two redundant interlocks which allow lifting arm operation only when the transfer car is at the end of its travel, one interlock is a position limit switch in the control circuit. The backup interlock is a mechanical latch device on the lifting arm that is opened by the car moving into position.

3. Transfer Car - Valve Open

   The interlock on the transfer tube valve permits transfer car operation only when the transfer tube valve position switch indicates that the valve is fully open. However, if the interlock fails in the permissive mode, the car can be started with the valve not fully open. The transfer car contacts the valve, stopping the transfer car movement. The car would stall without damage to the equipment or fuel. As soon as the valve is opened fully, the transfer car is freed.

4. Transfer Car - Lifting Arm

   The transfer car lifting arm interlock is primarily designed to protect the equipment from overload and possible damage if an attempt is made to move the car while the fuel container is in the vertical position. This interlock is redundant and can withstand a single failure. The basic interlock is a position limit switch in the control circuit. The backup interlock is a mechanical latch device that is opened by the weight of the fuel container when it is in the horizontal position

5. Lifting Arm - Refueling Machine

   The containment side lifting arm is interlocked with the refueling machine. Whenever the transfer car is located in the refueling canal, the lifting arm cannot be operated unless the refueling machine mast is in the fully retracted position or the refueling machine is over the core.

6. Lifting Arm - Fuel Building Fuel Handling Bridge Crane

   The Fuel Building side lifting arm is interlocked with the fuel handling bridge crane. The lifting arm cannot be operated unless the fuel handling bridge crane is not over the lifting arm area.

The Fuel Building fuel handling bridge crane includes the following safety features:

1. The fuel handling bridge crane controls are interlocked to prevent simultaneous operation of bridge drive and hoist.

2. Bridge drive operation is prevented except when the hoist is in the full up position.

3. An overload protection device is included on the hoist to limit the uplift force which could be applied to the fuel storage racks. The protection device limits the hoist load to the rated 2-ton-hoist capacity of 4000 lb and can withstand a single failure.

4. The design load on the hoist is the weight of one fuel assembly (1600 lb), weight of one failed fuel container (1000 lb), and the weight of the tool which gives it a total weight of approximately 3000 lb.

5. Restraining bars are provided on each truck to prevent the bridge from overturning.

All fuel handling tools and equipment handled over an open reactor vessel are designed to prevent inadvertent decoupling from machine hooks; i.e., lifting rigs are pinned to the machine hook, and safety latches are provided on hooks supporting tools.

Tools required for handling internal reactor components are designed with fail-safe features that prevent disengagement of the component in the event of operating mechanism malfunction. These safety features apply to the following tools:

1. Control rod Drive Shaft Unlatching Tool

   The air cylinders actuating the gripper mechanism are equipped with backup springs which close the gripper in the event of loss of air to the cylinder.

   Air-operated valves are equipped with safety locking rings to prevent inadvertent actuation.

2. Spent Fuel Handling Tool

   When the fingers are latched a pin is inserted into the operating handle and prevents inadvertent actuation. The tool weighs approximately 385 lb and is preoperationally tested at 125 percent the weight of one fuel assembly (1600 lb).

3. New Fuel Assembly Handling Tool

   When the fingers are latched a safety screw is screwed in, preventing inadvertent actuations. The tool weighs approximately 100 lb and is preoperationally tested 125 percent the weight of one fuel assembly (1600 lb).

The means have been developed to assure the safe handling of heavy loads at CPNPP. The consequences of dropping heavy loads were determined through a detailed analysis (Reference [17]), and the implementation of the CPNPP Control of Heavy Loads Program resulted in a significant reduction in the probability of a load drop occurrence.

For load handling systems where a potential load drop could affect the safe shutdown of the plant or result in unacceptable fuel damage, preventative measures have been developed and implemented as described in Reference [17]. Specifically, cranes which handle heavy loads in the area of spent fuel are designed with inherent safety features which prevent damage to fuel in the event of load drop. These inherent safety features are:

1. Containment Polar Crane

The Containment Polar Crane is used during the plant construction phase for lifts up to 475 tons (for handling the reactor vessel and steam generators) prior to its intended normal service.

The use of the crane during the construction phase does not imply any nuclear safety related condition.

During refueling or maintenance operations, the Containment Polar Crane handles a maximum critical load of 175 tons. The heaviest load expected to be lifted is the reactor vessel head assembly.

Due to the seismic design of the Containment Polar Crane, its structural integrity is sustained and its wheels are not dislodged from the track during an SSE.

The Containment Polar Cranes' main hoists have been equipped with single-failure-proof features. A detailed analysis of the features of the Containment Polar Crane has been made against the guidelines of NUREG-0554 "Single-Failure-Proof Cranes for Nuclear Power Plants." A summary of this analysis is provided in Reference [17]. The special safety features incorporated into the design of the main hoisting system of the Containment Polar Crane precludes a load drop accident by preventing a load drop in the event of a single failure in the hoisting or braking systems (Reference [17]). In addition, CPNPP design employs an integral steel missile shield (Unit 1 only) and a roll-away steel missile shield (Unit 2 only) over the reactor instead of a concrete-type shield. Therefore, no concrete structures are lifted in the vicinity of, or over, spent fuel.

2. Fuel Building Overhead Crane

The Fuel Building Overhead Crane cannot travel over the spent fuel pools. In addition, a spent fuel shipping cask, if dropped, cannot roll into the spent fuel pools.

However, to avoid placing undue restrictions on routine crane operations (including handling of spent fuel shipping casks) in the Fuel Building, the Fuel Building Overhead Crane's main hoist was equipped for spent fuel dry cask operations.

During refueling and dry cask storage operations, the heaviest loads handled by the Fuel Building Overhead Crane will be the spent fuel shipping cask and the loaded HI-TRAC transfer cask. The HI-TRAC transfer cask, with or without spent fuel, is considered a critical load. Heavy loads over an MPC with spent fuel, such as the MPC lid, HI-TRAC lid, Mating Device and HI-STORM lid, are considered critical loads. In addition, the transfer of an MPC with spent fuel between the HI-TRAC and the HI-STORM storage cask is a critical load.

The special safety features incorporated into the design of the main hoisting system of the Fuel Building Overhead Crane precludes a load drop accident by preventing a load drop in the event of a single failure in the hoisting or braking systems. Information regarding the Fuel Building Overhead Crane's compliance with the single-failure-proof provisions of Regulatory Guide 1.104 "Single-Failure-Proof Overhead Crane Handling Systems for Nuclear Power Plants" (Draft 3, Revision 1, October 1978) is provided in Reference [17].

The detailed evaluation of heavy load handling systems (Reference [17], safe load paths and areas, single-failure-proof cranes and hoists, and administrative control support the safe operation of CPNPP.

9.1.4.3.2 Seismic Considerations

The safety classification and seismic categories for all fuel handling and storage equipment are listed under Systems 33, 34, and 35 in Table 17A-1. These safety classes and seismic categories provide criteria for the seismic design of the various components. Safety Class 1 and Safety Class 2 equipment are designed to withstand the forces of the OBE and SSE. For normal conditions plus OBE loadings, the resulting stresses are limited to allowable working stresses as defined in the ASME B&PV Code, Section III, Appendix XVII, Subarticle XVII-2200 for normal and upset conditions. For normal conditions plus SSE loadings, the stresses are limited to within the allowable values given by ASME B&PV Code, Section III, Appendix XVII, Subarticle XVII-2110 for critical parts of the equipment which are required to maintain the capability of the equipment to perform its safety function. Permanent deformation is allowed for the loading combination which includes the SSE to the extent that there is no loss of safety function.

The Safety Class 3 fuel handling and storage equipment satisfies the Safety Class 1 and Safety Class 2 criteria for the SSE. Consideration is given to the OBE only insofar as failure of the Safety Class 3 equipment might adversely affect Safety class 1 or Safety Class 2 equipment.

The containment polar crane and fuel building overhead crane are listed in FSAR Table 17A-1 as seismic Category I. The safety related functional requirements are the retention of lifted loads by the main hoist during and after seismic events. The cranes are not otherwise required to remain functional after a seismic event. The loading combinations for the fuel building overhead crane include OBE and SSE with lifted load. Because of the extremely low probability of SSE in conjunction with a critical (heavy load, i.e., > 2150 lbs.) lifted load for the polar crane, the loading combinations include OBE with lifted load and SSE without lifted load.

The refueling gates are designated seismic category I in accordance with Reg. Guide 1.29 and Reg. Guide 1.13.

For non-nuclear-safety related equipment, design for the SSE is considered if failure might adversely affect a Safety Class 1, Safety Class 2, or Safety Class 3 component. Design for the OBE is considered if failure of the non-nuclear-safety component might adversely affect a Safety Class 1 or Safety Class 2 component.

Seismic design is not required for portable underwater lights. The use of lights containing mercury within the containment or fuel building are subject to precautions and limitations that assure nuclear safety is not compromised.

9.1.4.3.3 Containment Pressure Boundary Integrity

The fuel transfer tube which connects the refueling canal (inside the reactor containment) and the fuel storage area (outside the containment) is closed on the refueling canal side by a blind flange at all times except during refueling operations. Two seals are located around the periphery of the blind flange with leak-check provisions between them.

9.1.4.3.4 Radiation Shielding

During all phases of spent fuel transfer, the gamma dose rate at the surface of the water is 2.5 mrem/hr or less. This is accomplished by maintaining a minimum of 10 ft of water above the top of the fuel assembly during all handling operations.

The two fuel handling devices used to lift spent fuel assemblies are the refueling machine and the fuel handling bridge crane. The refueling machine contains positive stops which prevents the top of a fuel assembly from being raised to within a minimum of 10 ft of the normal water level in the refueling cavity.

The hoist on the fuel handling bridge crane and the containment fuel storage area crane moves spent fuel assemblies with a long-handled tool. Hoist travel and tool length likewise limit the maximum lift of a fuel assembly to within a minimum of 10 ft of the normal water level in the fuel storage area.

9.1.4.4 Inspection and Testing Requirements

The test and inspection requirements for the equipment in the FHS are as follows:

1. Fuel Handling Bridge Crane, Refueling Machine, RCC Changing Fixture, and New Fuel Elevator

   The minimum acceptable tests at the shopsite include the following:

   a. Hoist and cable are load-tested at 125 percent of the rated load.

   b. The equipment is assembled and checked for proper functional and running operation.

   The following maintenance and checkout tests are recommended to be performed prior to using the equipment:

   a. Visually inspect for loose or foreign parts. Keep free of dirt and grease.

   b. Lubricate wheels and exposed gears with proper lubricant.

   c. Inspect hoist cables for worn or broken strands.

   d. Visually inspect all limit switches and limit switch actuators for any sign of damaged or broken parts.

   e. Check the equipment for proper functional and running operation.

2. Head Lifting Rig and Internals Lifting Rig

   The minimum acceptable tests at the shopsite include the following:

   a. The rigs are load-tested to 125 percent of the rated load.

b. The rigs are assembled to ensure proper component fit-up.

The following maintenance and checkout tests are recommended to be performed prior to using the tools:

a. Visually inspect for loose or foreign parts or damaged surfaces.

b. Visually inspect all engagement surfaces and lubricate with proper lubricant.

c. On the internals lifting rig, check for the proper functioning of the engagement and protective rig operators.

3. New Fuel Assembly Handling Tool and Spent Fuel Assembly Handling Tool

The minimum acceptable tests at the shopsite include the following:

a. The tools are load-tested to 125 percent of the rated load.

b. The tools are assembled and checked for the proper functional operation.

The following maintenance and checkout tests are recommended to be performed prior to using the tools:

a. Visually inspect the tools for dirt, for loose hardware, and for any signs of damage such as nicks and burns.

b. Check the tools for proper functional operation.

4. Fuel Transfer System

The minimum acceptable tests at the shopsite include the following:

a. The system is assembled and checked for proper functional and running operation.

b. The fuel container is held in a 45-degree position for an unpowered condition.

The following maintenance and checkout tests are recommended to be performed prior to using the tools:

a. Visually inspect for loose or foreign parts. Keep free of dirt and grease.

b. Lubricate wheels and exposed gears with proper lubricant.

c. Visually inspect all limit switches and limit switch actuators for any sign of damaged or broken parts.

d. Check the system for proper functional and running operation.

5. Reactor Vessel Stud Tensioner

The minimum acceptable test at the shopsite is that the tensioner be assembled and checked for proper functional and running operation.

The following maintenance and checkout tests are recommended to be performed prior to using the equipment.

a. Visually inspect for loose or foreign parts.

b. Inspect hydraulic lines for wear or damage.

c. Check the hydraulic unit for proper pressurization and if any leaks occur at operating pressure.

6. Portable Underwater Lights

A visual inspection of lamp modules containing mercury is required prior to use.

9.1.4.5 Instrumentation Requirements

The control systems for the refueling and fuel handling machines and FTS are discussed in Subsection 9.1.4.2.3, Component Description. A discussion of additional electrical controls, such as the interlocks and the main hoist breaking system for the FHS, are discussed in Subsection 9.1.4.3.1, Safe Handling.

9.1.5 SPENT FUEL DRY CASK STORAGE

The CPNPP has established an Indpendent Spent Fuel Storage Installation (ISFSI) on an approximately 1.75 acre site located about 2900 feet east of the Unit 1 Containment Building and along the souther edge of the CPNPP site peninsula located within the Squaw Creek Reservoir.

The Spent Fuel Dry Cask Stoarge opertions at CPNPP will be conducted under a general license in accordance with Subpart K of 10 CFR Part 72. The general license issued by 10 CFR 72.210, “General license issued,” authorizes a 10 CFR Part 50 nuclear power plant license to store spent fuel at an onsite ISFSI. Subpart K of 10 CFR Part 72 also includes 10 CFR 72.212, “Conditions of general license issued under 10 CFR 72.210,” which requires the use of a dry cask storage system that is pre-approved by the Nuclear Regulatory Commission, as evidenced by its listing in 10 CFR 72.214.

The ISFSI site boundary has a security fence to establish a separate protected aea for the ISFSI. A nuisance fence is included on the perimeter of the outside isolation zone to minimize intrusion detection alarms.

The CPNPP ISFSI uses the Holtec HI-STORM 100S Version B vertical cask storage overpack and the Holtec MPC-32 multi-purpose canister (MPC), as described in the HI-STORM 100 Cask System FSAR [19] and approved by the Nuclear Regulatory Commission via the HI-STORM Certificate of Compliance No. 1014 [20]. The ISFSI concrete pad has a capacity for 84 vertical spent fuel storage casks.

REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, Design Bases for Protection Against Natural Phenomena.

2. 10 CFR Part 50, Appendix A, General Design Criterion 3, Fire Protection.

3. 10 CFR Part 50, Appendix A, General Design Criterion 4, Environmental and Missile Design Bases.

4. 10 CFR Part 50, Appendix A, General Design Criterion 5, Sharing of Structures, Systems, and Components.

5. 10 CFR Part 50, Appendix A, General Design Criterion 44, Cooling Water.

6. 10 CFR Part 50, Appendix A, General Design Criterion 45, Inspection of Cooling Water System.

7. 10 CFR Part 50, Appendix A, General Design Criterion 46, Testing of Cooling Water System.

8. 10 CFR Part 50, Appendix A, General Design Criterion 61, Fuel Storage and Handling and Radioactivity Control.

9. 10 CFR Part 50, Appendix A, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.

10. 10 CFR Part 50, Appendix A, General Design Criterion 63, Monitoring Fuel Waste and Storage.

11. NRC Regulatory Guide 1.13, Fuel Storage Facility Design Basis, March 1971.

12. NRC Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Revision 3, February 1976.

13. NRC Regulatory Guide 1.29, Seismic Design Classification, Revision 2, February 1976.

14. Branch Technical Position APCSB 3-1, Protection Against Postulated Piping Failure in Fluid Systems Outside Containment.

15. ANSI N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, 1973.

16. DELETED

17. Design Basis Document, DBD-ME-006, Control of Heavy Loads at Nuclear Plants (Incorporation by Reference), (Ref. 17 Originally “CPSES Final Response to NUREG-0612”, June 1983, as submitted by TXX-3659 dated June 8, 1983.)

18. License Amendment Request 00-05, Revision to Technical Specification Spent Fuel Assembly Storage Racks and Fuel Storage Capacity, Docket Nos. 50-445 and 50-446

a. TXU Electric letter logged TXX-00144, from C. L. Terry to USNRC dated October 4, 2000

b. TXU Electric letter logged TXX-01052, from C. L. Terry to USNRC dated March 21, 2001

c. TXU Electric letter logged TXX-01074, from C. L. Terry to USNRC dated April 30, 2001

d. TXU Electric letter logged TXX-01102, from C. L. Terry to USNRC dated June 18, 2001

e. TXU Electric letter logged TXX-01115, from C. L. Terry to USNRC dated June 27, 2001

f. TXU Electric letter logged TXX-01118, from C. L. Terry to USNRC dated July 18, 2001

g. TXU Electric letter logged TXX-01183, from C. L. Terry to USNRC dated January 22, 2002.

19. Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System, Revision No. 9, USNRC Docket No. 72-1014, Holtec Report No.: HI-2002444, February 13, 2010.

20. USNRC Certificate of Compliance No. 1014, Docket No. 72-1014, Amendment No. 7, for the HI-STORM 100 Cask System, December 28, 2009.

21. NUREG/CR-6407 Classification of Transportatin Packaging and Dry Spent Fuel Storage System Components according to Importance to Safety.

TABLE 9.1-1
SPENT FUEL POOL COOLING AND CLEANUP SYSTEM DECAY HEAT PARAMETERS (NOTE 1)

PARAMETER (NOTE 3)	MAX. DESIGN		MAX. SUMMER DESIGN		ABNORMAL MAX. DESIGN	
	MAX POOL	MIN POOL	MAX POOL	MIN POOL	MAX POOL	MIN POOL
Decay Heat Produced (x 10^6 BTU/hr) (Note 3)	52.0	9.09	23.28	6.08	62.92	6.6
SSI Temperature (°F)	94	-	102	-	102	-
Maximum SFP Temp (°F)	<150	<150	<150	<150	<212	<212
Time to boiling (hrs) (Note 3)	>3	-	>3	-	N/A	N/A

NOTES:

1. See Section 9.1.3.1.1 for the design conditions in this table.
2. Deleted.
3. Decay heat loads in “Max Pool” maximize temperatures and minimize time to boiling.
4. Assuming cooling is temporarily lost, the time to boiling is evaluated for the temperature rise from 150°F to 212°F.

TABLE 9.1-2
SPENT FUEL POOL COOLING AND CLEANUP SYSTEM CODE AND SAFETY CLASS REQUIREMENTS

Component	Safety Class	Code
Spent fuel pool cooling water pump	3	ASME III, Class 3
Refueling water purification pump	NNS	Mfrs. standard
Spent fuel pool skimmer pump	NNS	Mfrs. Standard
Spent fuel pool heat exchanger	3	ASME III, Class 3
Spent fuel pool demineralizer	3	ASME III, Class 3
Spent fuel pool filter	NNS	ASME VIII
Spent fuel pool skimmer filter	NNS	ASME VIII
Spent fuel pool suction screens	3	Mfrs. standard
Spent fuel pool skimmer	NNS	Mfrs. Standard
Spent fuel pool skimmer strainer	NNS	Mfrs. standard
Spent fuel pool cooling system pressure reduction orifice	3	ASME III, Class 3
Purification loop resin trap	3	ASME III, Class 3
Cask pit and transfer canal drain pump	NNS	Mfrs. standard
Piping and valves (nuclear)	3	ASME III, Class 3
Piping and valves (nuclear)	2	ASME III, Class 2
Piping and valve (non-nuclear)	NNS	ANSI B31.1
Refueling cavity purification pressure reduction orifice	NNS	Mfrs. standard

TABLE 9.1-3
TABLE DELETED

TABLE 9.1-4
NUMBER OF FUEL ASSEMBLIES IN SPENT FUEL POOLS BY REFUELING OUTAGE

(Sheet 1 of 2)

Outage Year	U1 Discharged Assemblies	U2 Discharged Assemblies	Total Discharged Assemblies Following Refueling	Remaining Storage Capability
1991	56	0	56	500
1992	61	0	118	440
1993	88	0	206	350
1994	0	83	289	267
1995	97	0	386	170
1996	85	96	567	715[(a)]
1997	0	77	644	638
1998	89	0	733	549
1999	89	89	911	371
2000	0	93	1004	799[(b)]
2001	93	0	1097	675
2002	89	90	1276	1706[(c)]
2003	0	81	1357	1598
2004	89	0	1446	1479
2005	85	101	1632	1231
2006	0	88	1720	1114
2007	89	0	1809	995
2008	93	89	1991	752
2009	0	93	2084	628
2010	93	0	2177	504
2011	92	92	2361	259

TABLE 9.1-4
NUMBER OF FUEL ASSEMBLIES IN SPENT FUEL POOLS BY REFUELING OUTAGE

(Sheet 2 of 2)

Outage Year	U1 Discharged Assemblies	U2 Discharged Assemblies	Total Discharged Assemblies Following Refueling	Remaining Storage Capability
2012	0	89	2162	524[(d)]

a) Use of Region II racks in SFP2 in a high density (2/4) storage configuration. (capacity of 1291).

b) Full Use of Region II racks in SFP2 in a high density (4/4) storage configuration (capacity of 2026).

c) Fuel Use of Region I and Region II racks in SFP1 and SFP2 (capacity of 3373).

d) Implementation of Dry Cask Storage at CPNPP to maintain SFP Storage capability.

9.2 WATER SYSTEMS

This section provides a discussion of each of the auxiliary water systems associated with the plant. These systems are: 1) Station Service Water System (SSWS); 2) Component Cooling Water System (CCWS); 3) Demineralized Water Makeup System; 4) Potable and Sanitary Water System; 5) Ultimate Heat Sink; 6) Condensate Storage Facilities.

9.2.1 STATION SERVICE WATER SYSTEM

9.2.1.1 Design Bases

The SSWS removes heat from the CCWS heat exchangers and from the emergency diesel generators, and supplies cooling water to the safety injection, centrifugal charging pump lube oil coolers and the containment spray pump bearing oil coolers. In conjunction with the CCWS (Section 9.2.2), the SSWS supplies cooling water to meet the plant cooling requirements during normal operation, shutdown, and during or after a postulated loss-of-coolant accident (LOCA) of either unit [5]. The required cooling water is taken from the safe shutdown impoundment (SSI), which is the ultimate heat sink and is designed in accordance with the guidelines in NRC Regulatory Guide 1.27 [11]. The SSWS also acts as a backup water supply for the Auxiliary Feedwater System if the Condensate Storage Tank is depleted.

The SSI contains a water supply for a minimum of 30 days of reactor decay heat removal, without outside makeup. The SSWS is designed to properly operate with water in the SSI at the lowest level during this period of time. (Sections 2.4.11.5 and 2.4.11.6)

The SSWS is designed to seismic Category I requirements [12].

The maximum allowable SSWS supply temperature is 117°F. The maximum calculated SSWS supply temperature is less than 116°F during LOCA conditions. See Section 9.2.5 for additional details. One system for each unit is provided as described in Section 9.2.1.2.

The SSWS is designed on the basis of the following:

1. Flow from one of the two redundant trains is continuously delivered to one of the two emergency diesel generators of each unit.

2. Flow is continuously delivered to one of two lube oil coolers per unit for the safety injection and centrifugal charging pump and two of four containment spray pump bearing oil coolers as shown on Figure 9.2-1 (Sheet 3 of 3).

3. Normal cooldown of one unit is accomplished with two SSWS pumps delivering water to two CCWS heat exchangers to cool the reactor coolant system (RCS) from 350°F to 140°F. The SSWS outlet temperature is maintained below 133°F.

4. An orderly shutdown can be achieved with one SSWS pump and one CCWS heat exchanger, although at a slower cooldown rate than outlined in Item 3 above.

5. One SSWS pump in conjunction with one CCWS heat exchanger provides one unit with enough cooling capacity for safe recovery after a LOCA (see Section 9.2.2.1).

The SSWS is vital to plant safety and is provided with redundant components so that no single failure denies cooling to equipment required for safe shutdown.

Equipment necessary for shutdown is supplied with emergency diesel generator power if normal and offsite power sources fail. In this event, at least one SSWS pump per unit is operative within 60 sec from the beginning of blackout (including emergency diesel generators starting time, sequencing delays, and pump starting time).

Chloride contamination of the CCWS is detrimental to the physical integrity of various stainless steel components supplied with component cooling water, such as reactor coolant pump heat exchangers, letdown heat exchangers, and so forth. In order to prevent such contamination through a tube leak in a CCWS heat exchanger, the component cooling water is at a higher pressure than the station service water.

9.2.1.2 System Description

9.2.1.2.1 General Description

Figure 9.2-1 (Sheets 1, 2 and 3) shows a schematic flow diagram of the SSWS. The major components of the system are described in Section 9.2.1.7.

The SSWS consists of two separate and independent full-capacity, safety-related trains. Cross connections between the two trains add operational flexibility to the SSWS.

The safety-related trains are redundant in that the components supplied by one train are sufficient to perform the minimum required safety functions. Two full capacity SSWS pumps and two full capacity supply and return headers are provided for each unit. Both pumps normally operate. This provides a continuous cooling water supply to the two redundant safety-related trains.

Controls are provided to allow remote operation from the Control Room or from the hot shutdown panel. A cross connection between Unit 1 and Unit 2 SSWS is provided.

The SSWS pump motors have couplers installed for connection of partial discharge monitoring equipment. The installation of diagnostic equipment is used for long term reliability monitoring and is not required for system operation.

During low water levels in the Squaw Creek Reservoir, the SSI is maintained at or above el. 769 ft, 6 in. by the safe shutdown dam, connecting canal. (Section 9.2.5)

A bleed line is provided from the Circulating Water System, at the intake structure, to the Service Water Intake Structure pump suction pit. See Section 9.2.5.2 for additional details.

The SSWS has a separate system which injects sodium hypochlorite and sodium bromide to control organic fouling. The effect of some fouling on heat transfer surfaces is considered in the design of the heat exchangers in the SSWS. A combined fouling factor of 0.0020 hr-ft^2-°F/BTU has been used on both the shell and tube side of the component cooling water heat exchanger for the evaluation of containment heat removal systems and CCW safety functions.

Bounding analyses are performed using the bounding design heat loads combined CCW heat exchanger fouling factor of 0.0020 hr-ft-°F/BTU and the maximum normal service water intake temperature of 102°F with no tube plugging. When service water intake temperatures are less than maximum, the fouling may exceed 0.0020 without any loss of quality or functional capability. The tube plugging margin is accounted for in the periodic fouling monitoring described below.

The quality of the SSI water is very similar to that of Squaw Creek Reservoir water. Stability and saturation index calculations have shown the reservoir water to be corrosive. In addition, to control the Asian clam population in the service water system and to help prevent the establishment of corrosion mechanisms within equipment, a non-oxidizing liquid biocide injection system is provided.

A chemical addition system is used to control corrosion and fouling in the Service Water System to protect the carbon steel pipe. A coordinated chemical treatment program with phosphate, organic phosphate, and copolymar is used in addition to the existing biocide treatments for this added corrosion protection.

Scale-forming tendencies are reduced as a result of the following:

1. Low water temperatures in the system (less than 110°F with a maximum of 133°F at the CCWS heat exchangers outlet for short periods during initial unit cooldown)

2. High water velocities are used in the system piping design.

Periodic monitoring determines when scaling is affecting design tube cleanliness factors. Safety-related systems serviced by SSWS are provided with redundant components so that if scaling affects tube cleanliness factors units are taken out of service during shutdown periods and cleaned.

This CCW heat exchanger fouling monitoring program (Section 9.2.1.8, item 5) ensures there is no loss of quality or functional capability between cleanings. In addition, more restrictive limits on system line ups, on SSI temperature and/or on CCW heat exchanger fouling are imposed on the operating (non-outage) unit as necessary during abnormal condition such as maintenance during outages.

9.2.1.2.2 Normal Operation

Normal operation of one unit requires the operation of both pumps associated with Train A and Train B. Operation of both SSWS trains continuously is desirable for corrosion control.

To insure reliability, these two pumps are connected to two separate emergency diesel buses.

9.2.1.2.3 Normal Unit Shutdown

For normal cooldown of one unit, two trains are used. The two trains provide the CCWS heat exchangers with the cooling capacity required to meet the design cooldown rate.

9.2.1.2.4 Operation During and After a LOCA

Train A or Train B provides enough cooling capacity for operation of the engineered safeguard features (ESF) required during the injection and recirculation phases of operation.

Normally during either phase, two trains are operating and both trains are aligned to cool down the unit under a LOCA.

During both the injection phase and the recirculation phase, only one train is required for safe recovery from the accident. However, both trains operate for more efficient energy removal from the Containment.

9.2.1.3 Leakage Detection and Control

All headers and major equipment are furnished with flow indication, which is used to detect abnormalities in the system flow rates. Abnormal flow conditions are annunciated in the Control Room. Abnormal leakage from systems located in plant buildings results in high sump levels, which are detected by level switches and also annunciated in the Control Room. A description of the means of detection of leakage through the CCWS heat exchangers is given in Sections 9.2.2.3 and 9.2.2.4. Upon detection of SSWS leakage, the leaking component can be identified by successive isolation of components until the leaking component is identified.

9.2.1.4 Safety Evaluation

The SSWS is designated ANSI N18.2 Safety Class 3 and designed to the requirements of seismic Category I [12], [15]. The SSWS pumps and related valves, operators, and instrumentation are located in the SSWS pumphouse which is a Category I structure, and is designed to withstand tornado effects. Piping between the SSWS pumphouse and the Auxiliary Building and from the Auxiliary Building to the SSI is buried underground with an amount of earth cover sufficient to protect the piping from tornado missiles. Other components are located inside the Auxiliary Building, which is a Category I structure and is designed against tornadoes. Pump motors, valve operators, and controls in the pumphouse are located above the highest postulated water level in the SSI. Valve operators and controls inside the Auxiliary Building also are located above the highest water level that might occur due to equipment failures within the building.

All of the previously mentioned conditions make the system capable of withstanding adverse environmental conditions, such as postulated earthquakes, tornadoes, and tornado missiles [1],[2]. (For details on seismic Category I structures, see Sections 3.2.1, 3.3, and 3.5.)

The SSWS is a moderate energy piping system. For a discussion of postulated pipe ruptures in this system, refer to Section 3.6 [14].

Pumps and pump motors inside the pumphouse are physically separated from each other by walls, as shown on Figures 1.2-45 and 1.2-46. These walls are designed to preclude coincident damage to redundant equipment in the event of a postulated pipe rupture, equipment failure, or missile generation.

Flooding is not considered possible because the floor of the pump compartments is at el. 796 ft, which is above the probable maximum water level of 791.3 ft. In the event of a pipe rupture

within the pump compartment, the water drains back to the SSI through large holes in the compartment floor.

The single-intake bay for all pumps, downstream of the intake channels and traveling screens, ensures flow to all pumps in the service water pumphouse.

The performance of all essential equipment can be monitored from the Control Room.

It is anticipated that no radioactive material will leak into the SSWS. The provisions for monitoring the SSWS for radioactive contamination are described in Section 11.5.

Environmental design conditions for seismic Category I self-actuated valves are described in Section 3.11.

9.2.1.5 Safety Implication Related to Sharing

There is no sharing of any safety-related component or safety-related function between the two units. The SSWSs include two redundant trains per unit [3]. Each train is separated from each other by two normally closed valves. GDC-5 is satisfied by a single locked closed valve in the cross-connection provided between units. The intake bay is provided with two traveling screens and associated screen wash equipment. There is no safety implication of this arrangement.

In the event of a loss of all essential cooling water on an operating unit, a cross-connect may be opened to provide backup cooling capability. One service water pump is capable of providing essential cooling to both units for this event by manually aligning and flow balancing. Therefore, there is no adverse safety implications of this back-up function (Ref. 17 and 18).

The cross-connects may also be opened under appropriate administrative controls for periodic flushing in accordance with Reference 19.

9.2.1.6 Single-Failure Analysis

A single-failure analysis is presented in Table 9.2-1.

This analysis shows that the failure or malfunction of any single active or passive component does not result in the loss of the minimum required safeguards functions [5], [11].

9.2.1.7 Components

The SSWS components safety classification is summarized in Section 3.2.2 and is indicated on the system's flow diagram on Figure 9.2-1. The major components in the SSWS are described as follows:

1. The four SSWS pumps are of the vertical centrifugal wet pit type, rated at 17,000 gpm at 140 ft of head.

2. The SSWS valves are either 316 stainless steel, carbon steel, nickel-plated carbon steel, or aluminum-bronze.

3. SSWS piping for sizes up to and including 3 1/2 in. is made of 304 or 316 stainless steel.

   Four inch diameter or larger piping is either carbon steel or 316 stainless steel.

   All joints and connections are welded, except at certain components where flanged connections are used to facilitate maintenance.

9.2.1.8 Testing and Inspection Requirements

Components in operation are interchanged periodically to enable testing and inspection [6],[7]. The standby components are inspected completely to find and correct incipient malfunctions. The inspection includes the following:

1. Pumps and Drive Motors

   Each pump is started and operated for sufficient time to insure its proper operation. Maintenance is provided as required.

2. Safety-Related Manual Valves

   At least once quarterly, Station Service Water System cross-connect valves XSW-0006, XSW-0007, XSW-0008, XSW-0028 and XSW-0029 will be stroked through their full range to insure that they are in operating condition.

3. Electrically Operated Safety-Related Valves

   All electrically operated safety-related valve control circuits are designed with the ability for on-line testing so that the availability and operational status can be readily determined.

4. Instrument and annunciator operation is checked periodically for proper operation and accuracy.

5. Heat Exchangers

   Safety-related heat exchangers are monitored periodically to establish effective capacity.

9.2.1.9 Instrumentation Requirements

A detailed description of instrumentation applications is given in Section 7.3.

9.2.1.10 Service Water Intake Structure Crane

9.2.1.10.1 Description

The Service Water Intake Structure (SWIS) Crane is located in the SWIS building (See Figure 1.2-46) and is used to install and maintain four service water pumps, three fire pumps and associated piping and equipment. The SWIS crane is an overhead traveling crane with an underhung 7.5 ton capacity hoist (see Figure 9.2-14). The main components of the SWIS crane are:

1. Runway beams with stops

2. Motor driven bridge and hoist

3. Runway conductors and collectors

4. Limit and overload switches

5. Pendant push button control station.

9.2.1.10.2 Design Criteria

The SWIS crane is designed to seismic category II (as defined in Appendix 17A and listed in Table 17A-1) and classified as a non-nuclear safety component.

9.2.1.10.3 Safety Evaluation

The SWIS crane is required to handle occasional non-critical loads and operate during normal operation of the plant including scheduled maintenance. If the Safe Shutdown Earthquake occurs, the load-bearing components, such as girders, wheels and runways are conservatively designed to remain in place with the wheels being prevented from leaving the tracks.

The separation and redundancy of the Station Service Water System ensures that, should an accidental load drop disable one train, at least two trains are still available for safe shutdown of both units.

9.2.2 COMPONENT COOLING WATER SYSTEM

9.2.2.1 Design Bases

The Component Cooling Water System (CCWS) provides an intermediate barrier between radioactive or potentially radioactive heat sources and the Station Service Water System (SSWS). It is designed to remove rejected heat from various plant components in a manner which precludes direct leakage of radioactive fluids into the Safe Shutdown Impoundment (SSI).

Leakage of the station service water into the component cooling water is detrimental to the physical integrity of various stainless steel components, such as reactor coolant pumps, heat exchangers, letdown heat exchangers and so forth, because of possible chloride-induced stress corrosion. To prevent such an occurrence, the CCWS is maintained at a higher pressure than the SSWS.

The CCWS is designed to supply cooling water for the components which are part of the Reactor Coolant System (RCS), Emergency Core Cooling System (ECCS), ESF systems, CVCS, Spent Fuel Pool Cooling and Cleanup System, Waste Processing System (WPS), ventilation system, and the Instrument Air System. The CCWS is required to operate according to the modes of operation as described in Table 9.2-4.

The CCWS is normally required to be operating during all phases of plant operation including startup, power operation, shutdown, refueling, and the injection and recirculation phases following a loss-of-coolant accident (LOCA). However, during outages, it may be necessary to

take both safeguards trains and/or the non-safeguards loop of CCW (see Section 9.2.2.2) out of service for maintenance. In this case, the common (shared) components are cooled by the operating unit. In this abnormal alignment, more restrictive limits may be imposed as described in Section 9.2.1.2.1.

The design cooldown rate, based on reducing the temperature of the reactor coolant from 350°F to 140°F will be achieved using two CCWS pumps and two CCWS heat exchangers. A slower but acceptable cooldown rate may be accomplished using only one CCWS pump and one CCWS heat exchanger. Under both conditions the CCWS supplies cooling water to its RHR heat exchanger at a maximum temperature of 122°F during its early part of cooldown. The supply water temperature will gradually return to 108°F towards the end of cooldown.

Because the system is required to perform its safety function during the short-term and long-term plant accident conditions, the safety related passive components as well as the active components are designed to meet the single failure criteria [12]. An analysis of postulated cracks in moderate energy systems is in Section 3.6. All ANSI Class 2 and 3 components of the CCWS are designed to meet seismic Category I requirements. Equipment that is necessary for shutdown and equipment that is required to mitigate the effects of an accident is supplied with emergency diesel power, should normal and offsite power sources fail.

The CCWS is designed in accordance with the following criteria, regulatory guides, and codes:

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants GDC 2 [1], 4 [2], 5 [3], 44 [5], 45 [6], 46 [7], 56 [8], and 57 [9]

2. NRC Regulatory Guides 1.26 [10], 1.29 [12], and 1.48 [13]

3. Codes

    a. ASME B&PV Code, Section III, Nuclear Power Plant Components

    b. ASME B&PV Code, Section XI, Inservice Inspection

    c. ANSI B31.1.0 Power Piping

    d. TEMA, Standard of Tubular Exchanger Manufacturers Association

    e. ASME OM Code

4. ANSI N18.2-1973 [15]

9.2.2.2 System Description

One system per unit is provided and each consists of three subsystems. Each system contains a Train A safeguards loop, a Train B safeguards loop, and a non-safeguards loop. The design allows common components, such as control room A/C condensers, to be supplied by either Unit 1 or Unit 2 CCW or a combination of the two. The two systems operate independently but can be cross-connected if needed.

Figure 9.2-3 presents the flow diagram of the Unit 1 system.

Each system consists of two 100-percent-capacity CCWS pumps and heat exchangers, with one pump and one heat exchanger operating and one pump and one heat exchanger on standby. Each system consists of two separate, redundant and independent full-capacity safeguards loops to service the engineered safeguards components and a nonredundant nonsafeguards loop with ANSI safety Class 3 and non-safety-class portions.

The CCW pump motors have couplers installed for connection of partial discharge monitoring equipment. The installation of diagnostic equipment is used for long term reliability monitoring and is not required for system operation.

During unit power operation, components in the same safeguards loop can be inoperable at any one time, provided that the other safeguards loop remains operative.

The equipment comprising the previously described loops is listed in the following paragraphs.

1. Each of the two safeguards loops consists of cooling to the following:

    a. Two Containment spray pump seal coolers

    b. One RHR pump seal cooler

    c. One RHR heat exchanger

    d. One Containment spray heat exchanger

    e. One Chilled Water System condenser (nuclear)

    f. Two Control Room air-conditioning condensers (common)

    g. One UPS air-conditioning condenser (common)

    In addition, train-A of the safeguards loop supplies cooling to the reactor coolant post accident sampling system sample cooler.

2. The nonsafeguards loop consists of cooling to the following:

    a. Two catalytic recombiner heat exchangers (common)

    b. Two waste gas compressor seal water coolers (common)

    c. One positive displacement charging pump hydraulic coupling oil cooler and lube oil cooler

    d. One letdown heat exchanger (CVCS letdown cooling)

    e. One seal water heat exchanger (CVCS reactor coolant pumps seals)

    f. Two spent fuel pool heat exchangers (common)

g. One boron recycle evaporator package consisting of one distillate cooler, one evaporator condenser, and one vent condenser (common)

h. One waste evaporator package consisting of one distillate cooler, one evaporator condenser, and one vent condenser (common)

i. One floor drain evaporator package consisting of one distillate cooler, one evaporator condenser, and one vent condenser (common)

j. Four reactor coolant pump packages, each consisting of one upper bearing lube oil cooler, two motor air coolers, one lower bearing lube oil cooler, and one thermal barrier cooler (located inside Containment)

k. One excess letdown heat exchanger and one reactor coolant drain tank heat exchanger (CVCS equipment located inside Containment)

l. Four ventilation chillers (common)

m. One letdown chiller package condenser

n. Two rotary instrument air compressor packages, each consisting of one intercooler, one aftercooler and one oil cooler.

o. Deleted

p. Process sample coolers

Items c to k inclusive above are ANSI Safety Class 3 components; items a, b and l to p inclusive above are non-safety-class-components. Common components are designed to be cooled by either Unit 1 or 2 CCW with the demand being shared about equally between the two.

The two safeguards loops per unit are redundant in that the components supplied with cooling water by either of the safeguards loops can perform the minimum required safeguards functions.

Two branches from the nonsafeguards loop penetrate the Containment; one supplies cooling water to the reactor coolant pumps, and the other supplies the excess letdown heat exchanger and reactor coolant drain tank heat exchanger. A third branch supplies the spent fuel pool heat exchangers.

The Unit 1 and 2 CCW are normally separated from each other by at least one closed valve.

Component cooling water flow to the non-safeguards loop is terminated by means of redundant, motor-operated valves upon initiation of containment spray and phase B containment isolation.

The safeguards loops are isolated either by locked-closed valves or by redundant, motor-operated valves actuated by a P or phase B isolation signal, or upon a surge tank Empty Level signal.

A partitioned surge tank for each unit vented to the atmosphere is provided with a separate chamber and separate piping for each safeguards loop to accommodate expansion, contraction,

and supply makeup water from a redundant Safety Class 3 source. The CCWS surge tank is located at an elevation which assures adequate net positive suction head (NPSH) to the CCWS pumps, and is the highest point in the system. The surge tank partition has one opening at the top, and another above the “Empty Level” set point to equalize the levels of both chambers. The partition is designed to maintain its integrity with one side of the surge tank empty.

The CCWS is initially filled with demineralized water to which a suitable corrosion inhibitor is added. Water chemistry is described in Table 9.2-3. A pot-type, manual, chemical addition system is provided for makeup during system operation.

To preclude discharging corrosion-inhibited component cooling and chilled water to the environment, one drain tank for each unit is provided in the Safeguards Building to collect component cooling water from piping and equipment being drained and from the discharge of most of the CCW thermal relief valves. Drainage flows to the drain tank by gravity. Two 100-percent-capacity CCWS drain pumps are provided per unit. Discharge lines from some CCW thermal relief valves and overflow lines from the CCWS surge tank are routed to the floor drains.

One CCWS drain pump from each unit is used to pump the drains to the truck discharge connection, or to the Wastewater Management System (see Section 9.2.8), depending on the quality and quantity of the water collected.

One Containment CCWS drain tank for each unit is provided to collect component cooling water from components in the Containment. Two 100- percent-capacity Containment CCWS drain pumps are provided per unit. One pump for each unit is used to pump the drains back to the main drain tank in the Safeguards Building. One Containment penetration is provided for the drain’s line.

The CCWS water chemistry (see Table 9.2-3) is controlled via a filter/demineralizer/chemical addition skid (one per Unit) and by the addition of chemicals via the chemical addition tank. Samples are periodically taken from sample taps to verify that water chemistry is as required. If dilution is required, water is bled off to the drain tank through one of the existing drain connections. Noninhibited water is added through the CCWS surge tank.

9.2.2.2.1 Modes of Operation

1. Normal Operation

   During normal operation, each system has one CCWS pump and one CCWS heat exchanger in operation. A second CCWS pump along with its heat exchanger is on standby (redundant) and aligned to replace the pump and heat exchanger in service. If the CCWS pump in operation trips, the redundant CCWS pump automatically starts and is operative within 60 sec.

   To ensure reliability, these two pumps are connected to two separate, emergency diesel buses. Each safeguards loop is supplied with emergency power from a separate diesel generator. Component cooling water is normally supplied through the operating heat exchanger to both the safeguards loops and to the nonsafeguards loops, including one spent fuel pool heat exchanger and two ventilation condensers. Component cooling

water is not supplied to RHR and Containment spray heat exchangers during normal operation.

Manual Valves CC-0107, CC-0109, CC-0157, and CC-0158 have modified discs which have been drilled to serve as flow restriction orifices during MODES 1 and 2. These valves are required to be closed to provide acceptable CCW flow balancing for Design Basis Accidents.

2. Normal Unit Shutdown

The cooldown phase requiring use of the RHR System starts approximately four hrs after reactor shutdown. At this time two CCWS heat exchangers and two CCWS pumps are aligned to all loops of the subsystem serving the unit being cooled down. Component cooling water is supplied to both RHR heat exchangers.

Manual Valves CC-0109 and CC-0157, which provide a flow limiting function in MODES 1, 2, and 3, may be opened in MODE 3 at or below 400°F as needed to support RHR cooldown in MODE 4, 5 and 6. Manual Valves CC-0107 and CC-0158, which provide a flow limiting function in MODES 1 and 2, may be opin MODE 4, 5, and 6.

3. Operation During and After LOCA

Component cooling water is continuously supplied to both safeguards loops so that any of the redundant safeguards pumps can be immediately started. Following Containment spray initiation, the nonsafeguards loops are isolated and do not receive component cooling water.

The safeguards loops are isolated from each other so that each loop is supplied by a separate CCWS heat exchanger. Component cooling water is supplied to the RHR heat exchanger and the Containment spray heat exchangers of both loops, as well as to the Control Room air-conditioning condensers and the Chilled Water System condenser, although one loop is sufficient to maintain the Containment temperature and pressure within the design limits and to supply sufficient cooling for condensers. A single failure of any component does not prevent the system from supplying water to one safeguards loop, as described in Table 9.2-5.

Manual Valves CC-0107, CC-0109, CC-0157, and CC-0158, which provide a flow limiting function in the short term after a Design Basis Accident (e.g. LOCA), may be opened to accelerate cooldown after accident heat loads have sufficiently decayed if desired and if the valve locations are accessible.

4. Failure-Mode Analysis

A single-failure analysis is presented in Table 9.2-5 and shows that the failure or malfunction of any single active or passive component does not prevent fulfillment of the safeguards functions.

9.2.2.2.2 Equipment Design Bases

The following are the design bases for the major components of the CCWS. A more detailed equipment listing and description showing design, performance, and material data is found in Table 9.2-2.

1. CCWS Heat Exchangers

Two 100-percent-capacity, single-pass shell and straight-tube heat exchangers are provided for heat transfer between the CCWS and the SSWS. Service water, which has a greater tendency to foul heat transfer surfaces than component cooling water, flows through the straight tubes to facilitate cleaning. As stated under design bases, since service water can contain chloride which is detrimental to stainless steel components, the component cooling water pressure in the heat exchanger is higher than the service water pressure to prevent service water in-leakage.

The heat exchangers have straight tubes and fixed tubesheets, with tubes rolled into the tubesheets and with channels and removable channel covers (TEMA Type CEN). They are designated ANSI Safety Class 3, seismic Category I. Both tubes and shell sides of the heat exchangers are designed to the requirements of the ASME B&PV Code, Section III, Code Class 3, and to TEMA Class R standards for items not specifically covered by the ASME B&PV Code.

2. CCWS Pumps

Two 100-percent-capacity pumps are provided. The pumps are horizontal centrifugal, double suction split case.

They are designated as ANSI Safety Class 3, seismic Category I and are designed to the requirements of the ASME B&PV Code, Section III, Code Class 3.

3. CCWS Surge Tank

The CCWS surge tank accommodates surges resulting from component cooling water thermal expansion and contraction and collects any water which may leak into the system from components which are being cooled. In the event that these surges are extreme, alarms will be triggered that require manual action by the operator. However, this will not jeopardize the safety function of the CCWS system. The surge tank also contains a supply of water to provide makeup as required until a leaking cooling line can be isolated. The CCWS surge tank is the highest point of the system to facilitate proper filling and venting of the CCWS system and to ensure adequate net positive suction head (NPSH) for the pumps during normal operation and accident conditions.

Two horizontal cylindrical surge tanks are provided (one per unit). Each surge tank has a vertical partition normal to centerline to provide two compartments, one compartment for each CCWS pump. The NPSH for the CCWS pumps are summarized in Table 9.2-2

The tank components are designated as ANSI Safety Class 3, seismic Category I, and are designed to the requirements of the ASME B&PV Code, Section III, Code Class 3.

4. Piping and Valves

The principal branches of the CCWS and their respective pipe categories are summarized in Table 9.2-2. Valves of the CCWS are made of carbon steel. Relief valves are made of either carbon steel or stainless steel. Thermal relief valves are secondary equipment overpressure devices, positioned near or on the component to relieve the volumetric expansion which occurs if the cooling water lines to the component are isolated and the water temperature rises. The thermal relief valve set pressures are less than or equal to 110% of the system or component design pressure.

The CPNPP technical specifications include a surveillance requirement to verify that each automatic valve serving safety related equipment actuates to its correct position on all applicable test signal. Periodic test procedures will be written to satisfy these surveillance requirements.

9.2.2.3 Safety Evaluation

Each component cooling water system is comprised of two full-capacity safeguards loops and one nonsafeguards loop. During emergency operation, the nonsafeguards loop is isolated from the safeguards loops by two automatic valves. The two safeguards loops are separated from each other either by redundant, remotely operated valves or by locked-closed valves.

The nonsafeguards loop penetrates the Containment to supply the reactor coolant pumps, excess letdown heat exchanger, and reactor coolant drain tank heat exchanger. Containment isolation valves are supplied at each penetration in accordance with the requirements of GDC 56 and 57. (See Section 6.2.4.)

The check valves supplied on the CCWS supply lines inside the Containment prevent reverse flow in case of pipe rupture.

The performance of all essential equipment can be monitored from the Control Room. Low flow, high temperature, and high radiation level, which are indicative of system malfunction, are annunciated in the Control Room. System pressure indication is provided in the Control Room.

Leakage from any system being cooled into the CCWS can be detected as an increase in the level of the CCWS surge tank or as an increase in system radiation level when the system being cooled is contaminated. Details of the radiation monitoring equipment are given in Section 11.5. Leakage from the CCWS to the SSWS or to the atmosphere is detected as a decrease in the CCWS surge tank level.

The ability to detect leakages to or from the CCWS permits the operator to take appropriate action before degradation of both boundaries between normally radioactive fluid and the SSWS can occur.

The partition in the surge tank below its equalizing opening provides separate surge volumes for each safeguards loop. If one of them develops a leak and is taken out of service, the operation of the other loop is unaffected. If a leak develops in the nonsafeguards loop, it can be isolated from the safeguards loops and their operation left unaffected. The partition is designed to maintain its integrity with one side of the surge tank empty.

In the unlikely event that leakage from the system exceeds the makeup rate such that the level in the surge tank drops faster than the operator can respond, each safeguards loop and the non-safeguards loop are automatically isolated from one another to prevent loss of inventory from both safeguards loops.

All major components of the system, piping, and its appurtenances are located inside the seismic Category I structure. Each CCWS pump is located in an individual, compartment and at an elevation above the highest water level that can occur as a result of equipment failures within the building.

Manual Valves CC-0107, CC-0109, CC-0157, and CC-0158 have modified discs which have been drilled to serve as flow restriction orifices. These valves are closed to provide acceptable CCW flow balancing for Design Basis Accidents to limit heat addition to CCW. These valves are not required to be opened to mitigate a Design Basis Accident (e.g. LOCA); however, they may be opened to accelerate cooldown after accident heat loads have sufficiently decayed if desired and if the valve locations are accessible. However, CC-0109 and CC-0157 are classified as Active Valves (see Table 3.9B-10) for an open safety function because they may be required to be opened to satisfy the functional requirements for cooldown of BTP RSB 5-1 (see Appendix 5A).

The non-safety radiation monitors and their sample piping on the safeguards loops are seismically supported to ensure their failure does not adversely affect the CCWS.

The previously mentioned conditions and the design of the CCWS as seismic Category I make the system capable of withstanding adverse environmental occurrences such as postulated earthquakes, tornadoes, and tornado missiles (for details on seismic Category I structures, see Sections 3.2, 3.3, and 3.5).

The compartment walls around each pump preclude coincident damage to redundant equipment in the event of a postulated pipe rupture or an equipment failure with consequent pipe whip, jet impingement, and missile generation.

CCW containment isolation valves in the return line from the RCPs thermal barrier close automatically in the event of a RCP thermal barrier tube break. CCW flow from the thermal barrier of all four RCPs is stopped automatically whenever outlet flow or temperature reaches a high value, indicative of a thermal barrier break. Failure of the nonsafeguards loop (loss of CCW) has the following effect upon the reactor coolant pumps:

1. The chemical and volume control system continues to provide seal injection water to the RCPs; the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling.

2. The motor winding temperature will increase as a result of the loss of CCW to the motor air cooler.

3. Lube oil temperature will increase, with a corresponding rise in bearing metal temperature, as a result of the loss of CCW to the motor bearing oil coolers. Testing, performed by Westinghouse, has shown that the manufacturer's recommended maximum bearing operating temperature will be reached in approximately 10 minutes. Therefore, the RCPs will incur no damage with a CCW flow interruption of 10 minutes.

4. No circuitry is provided to directly trip the RCPs on loss of component cooling water.

The following component cooling water instrumentation is provided:

1. Cooling water flow and temperature for all components associated with each RCP are indicated in the Control Room.

    a. RCP motor bearing high temperature alarms for each RCP are provided on the plant computer.

    b. Temperature indicator and alarm for the CCWS return line from each RCP are located on the main control board.

    c. CCWS flow indicator and alarm for each RCP oil cooler are located on the main control board.

    d. CCWS isolation valve monitor lights, indicating valve closure, are also located on the main control board.

2. Low cooling water flows are annunciated in the Control Room.

    Westinghouse has conducted a human engineering analysis of a loss of CCW to the RCPs, considering (1) the instrumentation provided, (2) the psychological stress induced on the average trained operator, and (3) the response time required of the operator to trip the reactor and stop the RCPs. It was found that the stress is much less than that induced by a LOCA (which would cause a response time delay of one minute), and that the operator actions required are not complicated and are a direct logical result of the event symptoms, as alarmed and indicated. In view of these factors, 10 minutes is a conservative and appropriate operator response time for this event during normal operation. In addition, since testing performed by Westinghouse has demonstrated that the RCPs are capable of operation for a 10-minute CCW flow interruption to the oil coolers without damage to the pumps, a period of ten minutes is adequate for the operator to take corrective actions or to trip the pumps, if necessary.

    During startup, cooldown and cold shutdown, two CCW pumps are normally used to supply all CCW requirements for one unit, including two RHR pumps. Failure of one of the CCW pumps or a single incorrect valve closure or a single moderate energy line crack would cause a temporary reduction of CCW flow to individual components while the CCWS is realigned. As discussed above, all low flow alarms and control for realignment are contained in the Control Room, and the time period for operator response would be no more than 10 minutes.

    Depending on the type of failure, the reduced flow rate would be no less than 50% of the normal flow rate. The RHR pumps would continue to operate as required under the circumstances. At no time would there be a cessation of CCW flow to either of the RHR pumps except in the case of a pipe crack in certain lines supplying an RHR pump. The realignment in this case would cause complete loss of CCW flow to one of the RHR pumps; the other RHR pump would continue to be supplied with CCW and cooldown would be accomplished using the single RHR pump. Cold shutdown may be achieved using only one CCW pump, if the other pump is not available.

Cold shutdown may be maintained using only one operating CCW pump with the other pump on standby. Failure of the operating pump would result in the automatic start of the standby pump within 30 seconds. Loss of CCW flow to the RHR pumps for this period of time can be tolerated.

During safeguards operation, the CCWS is train aligned. At this time, one CCW pump supplies one safeguards train, including one RHR pump. The other CCW pump supplies the other safeguards train, including the other RHR pump. Loss of any part of one safeguards train can be tolerated.

9.2.2.3.1 Safety Implications Related to Sharing

The CCWS includes two redundant CCWS pumps, two redundant CCWS heat exchangers, two redundant safeguards loops, and one nonsafeguards loop. The interconnections between Unit 1 safeguards loops and Unit 2 safeguards loops are blocked by at least one locked-closed isolation valve. Component cooling water for control room air-conditioning and UPS air-conditioning may be provided by either unit as these are shared systems (see FSAR Section 9.4.1 and 9.4C.8, respectively). There is no sharing of any other safeguards component or safeguards function between the two units.

9.2.2.3.2 Leakage Detection and Control

Leakage from any component being cooled into the CCWS can be detected as an increase in the level of the CCWS surge tank or as an increase in CCWS radiation level when the component being cooled is contaminated. Leakage from the CCWS to the SSWS or to the atmosphere is detectable by a decrease in CCWS surge tank level.

Detection of leakage in the CCWS surge tank is followed by the manual operation of either the feed system from demineralized water or the feed system from reactor makeup water, or the alarmed automatic actuation of the emergency feed system from reactor makeup water. Each CCWS surge tank level is recorded in the Control Room to facilitate leakage detection.

The CCWS surge tank contains water to provide a continuous component cooling water supply until a leaking CCWS loop can be taken out of service, with flow transferred to the other CCWS loop. The partition in the surge tank provides two compartments for separate surge volumes for each safeguards redundant loop. If one loop develops a leak and is taken out of service, the operation of the other loop is unaffected. A leaking safeguards redundant loop can be identified by isolating each loop of the system with the water level below the equalizing opening, and observing the water level in the related surge tank compartment volume. If the level stops decreasing when the safeguards loop is isolated, the source of leakage must be, by elimination of sources, in the nonsafeguards loop. In this case, if outside the Containment, the leaking component can be identified by visual inspection. A leaking component inside the Containment is identified by abnormal flow indication from that component in the Control Room.

9.2.2.4 Tests and Inspection

Prior to installation in the CCWS, each component is inspected and cleaned.

Preoperational testing consists of calibrating the instruments, testing the automatic controls for actuation of the proper set points, and checking the operability and limits of alarm functions.

Periodic chemical examination of the component cooling water is made for pH, chloride content, and corrosion inhibitor content; manual adjustment is made if required. Sufficient instruments are provided to monitor system performance. Periodic visual inspection and preventive maintenance can be conducted as necessary without interruption of the CCWS operation.

9.2.2.5 Instrumentation and Control Requirements

9.2.2.5.1 General System Monitoring

Each power-operated valve is supplied with a control switch and a position indicating light on the control board in the Control Room. Radiation monitors are provided with readout in the Control Room.

Separate switches and actuation circuitry are provided for redundant components which are physically and electrically separated from one another.

9.2.2.5.2 Instrumentation

1. Pressure

   Pressure indication for each CCWS pump discharge is provided both locally and in the Control Room. Pressure indication is provided locally for CCWS pump suction.

2. Temperature

   a. CCWS heat exchanger outlet temperature is indicated in the Control Room. A high outlet-temperature alarm is also provided in the Control Room.

   b. Containment spray heat exchanger and RHR heat exchanger outlet temperatures are indicated in the Control Room.

   c. Local temperature indication is provided for all components, except for the process sample coolers and the components inside the Containment. The process sample coolers have temperature test wells, and the components inside the Containment have indication in the Control Room.

3. Flow

   a. Flow indication is provided in the Control Room for component cooling water recirculation, component cooling water main flow to the loops, RHR heat exchangers, and Containment spray heat exchangers, in addition to all components inside the Containment (reactor coolant pumps accessories, excess letdown heat exchanger, and reactor coolant drain tank heat exchanger). The component cooling water flow is also recorded. Sufficient flow indication at all CCWS components is provided for flow balancing the system.

   b. Low flow alarms are provided for all CCWS components except for the process sample coolers, the control room air- conditioning condensers, the safety chilled water system condensers, and the ventilation chillers. The ventilation chillers have high-flow alarms.

4. Level

CCWS surge tank level indication is provided for each compartment locally and in the Control Room. It is also alarmed and recorded. CCWS drain tanks level indication is provided locally and is alarmed.

9.2.2.5.3 Control

1. CCWS Pumps and Remote-Operated Valves

a. All pumps and remote-operated valves are operated from the Control Room.

b. All remote-operated valves are supplied with open-close position indicating lights in the Control Room. Both lights are on when the valve is in the intermediate position.

c. CCWS pumps for both trains A & B have control switches located on the hot shutdown panel (HSP). In addition, CCWS Pump Train A transfer switch (CR-HSP) is located on the shutdown transfer panel, while CCWS pump train B transfer switch is located on the hot shutdown panel. Local control position alarms are provided in the Control Room.

d. If a low CCWS pressure signal is received, the redundant CCWS pump and its associated SSWS pump automatically start. The CCWS pump and the SSWS pumps are train associated pumps.

e. If a SSWS pump automatic start on low SSWS pressure occurs, the corresponding CCWS pump is automatically started to ensure correct alignment of the safeguard train.

f. During blackout conditions, CCW pumps are automatically loaded onto the emergency diesel generator within 30 seconds of blackout.

2. Makeup

Makeup to the CCWS Surge tank is normally obtained from the Demineralized Water Storage Tank (DWST) or the Reactor Makeup Water Storage Tank (RMWST) by remote manual operation of valves for either surge tank compartment. The operator is informed of the need for makeup by a low-level alarm for each compartment. The demineralized water pumps are used to transfer makeup water from the demineralized water system. If the alarm signal does not reach the operator because of equipment failure or operator inattention, makeup is automatically supplied by the emergency makeup system from the Reactor Makeup Water Storage Tank (RMWST). A low-low-level signal opens valves for either surge tank compartment and actuates a low-low-level alarm in the Control Room. The low-low-level actuated valves close automatically when the water in the surge tank reaches the high level.

Makeup can also be supplied by manual operation of level control valves, should the Instrument Air System become unavailable.

3. CCWS Pump Recirculation

If the CCWS pump flow falls below the required minimum, the CCWS recirculation valve opens automatically. Once the component cooling water flow increases above the required value, the valve closes automatically. The recirculation valve is also automatically closed upon receipt of an “S” signal.

4. Safeguards Chilled Water Condenser/Nuclear

The cooling of the chilled water is based on the condenser refrigerant pressure. A pressure controller on the refrigerant side of the condenser adjusts the CCW regulating valve on the condenser outlet piping. This valve is air operated with accumulator tank backup. The accumulator tank is sized to supply air for a minimum of 30 minutes after an instrument air header low pressure condition. The valve is handwheel equipped for manual operation by a control room operator when instrument air is exhausted and CCW temperature is below 70 degrees F.

5. Evaporator Flow Control

Component cooling water to the evaporator condensers of the various evaporator packages is modulated within the evaporator package to meet demand.

6. Ventilation Chillers and Letdown Chillers

Flow to the ventilation chillers is stopped automatically when it reaches an abnormally high value indicative of leak or system malfunction. In addition, the CCWS inlet and outlet valves to the ventilation chiller and letdown chiller condensers are closed automatically on an “S” signal.

The cooling of the ventilation chiller is based on the condenser refrigerant pressure. A pressure indicating controller (PIC) on the refrigerant side of the condenser adjusts the CCW regulating valve (200-2000 gpm) to the condenser and inversely the CCW bypass valve (1800 - 0 gpm) for a total flow around/across the condenser of 2000 gpm.

7. Thermal Barrier Cooler

RCP thermal barrier CCW outlet flow and temperature instrumentation provides Class 1E, redundant and diverse signals for Class 1E indication in the main control room and automatic closure of CCW containment isolation valves upon sensing RCP thermal barrier tube break. A Train A flow transmitter in the return from each thermal barrier (FT-4678, FT-4682, FT-4686, or FT-4690) will automatically close Train A Containment Isolation Valve HV-4709 on high flow. A Train B temperature element in the return from each thermal barrier (TE-4691, TE-4692, TE-4693, or TE-4694) will automatically close Train B Containment Isolation Valve HV-4696 on high temperature (See Figure 9.2-3).

In addition, flow from the thermal barrier of any RCP is stopped automatically with control grade valves (TV-4691, TV-4692, TV-4693, or TV-4694) whenever the outlet temperature (TE-4691, TE-4692, TE-4693, or TE-4694) reaches an abnormally high value indicative of a thermal barrier break. These control valves provide an additional level of redundancy

and diversity of protection from an intersystem LOCA. Control grade alarms are provided for outlet high flow and temperature (See Figure 9.2-3).

8. Drain Tank Pumps

CCWS drain tank pumps are interlocked to stop on low drain tank level.

9. S Signal

When an S signal occurs, the following operations take place:

a. Both CCWS pumps are started.

b. The Containment CCWS drain tank pumps stop.

c. The pump recirculation valves close.

d. The RHR heat exchanger CCWS isolation valves open fully then partially close to a preset throttle position.

e. CCWS Containment phase “A” isolation valves close as described in Section 6.2.4.

f. The ventilation chiller condenser’s and letdown chiller condenser’s isolation valves close.

g. The Primary Sampling System CCW isolation valves close as listed in Table 7.3-4.

10. P Signal

When a P signal occurs the following operations take place.

a. The RHR heat exchanger CCWS isolation valves open fully then partially close.

b. The Containment spray heat exchanger CCWS isolation valves open fully then partially close to a preset throttle position.

c. CCWS containment Phase “B” isolation valves close as described in Section 6.2.4.

d. The safeguards loop isolation valves close.

e. The non-safeguards loop isolation valves close.

9.2.2.5.4 Radiation Monitoring

Radiation monitors are provided on the return headers in each loop of the CCWS. Any of these monitors can trigger a high-activity-level alarm signal in the Control Room. A temperature switch is provided to trip the monitor sample pump if the temperature exceeds 120°F to protect the detector.

A radiation monitor is also provided in the common discharge line of the CCW Drain Tanks, Auxiliary Building Sumps 3 and 11, and the Diesel Generator sumps. A high radiation signal or radiation monitor failure diverts the discharge from the LVW System to the Cocurrent Waste System.

9.2.3 DEMINERALIZED AND REACTOR MAKEUP WATER SYSTEM

9.2.3.1 Design Bases

The design of the Demineralized and Reactor Makeup Water System is in accordance with the following criteria: 10CFR50, Appendix-A, General Design Criteria (GDC) 1, 2, 3, 4, 5, 52, 54, and 56; NRC Regulatory Guide 1.26 and 1.29; and Branch Technical Position APCSB 3-1. The related electrical systems are designed to comply with the requirements of regulatory guides, standards and other documents as described in Section 8.1.4. The related Containment Penetrations are designed to comply with the requirements described in Section 6.2.4. The Demineralized and Reactor Makeup Water System is a single system designed to provide an adequate supply of deaerated demineralized water of reactor coolant purity to other systems as makeup and to provide other plant demineralized water requirements for both units of the CPNPP.

The Demineralized Water Storage Tank (DWST) receives water from the Water Treatment System. When treated water enters the DWST, the quality of water in the tank is consistent with the water chemistry listed in Table 9.2-9 for demineralized water saturated with oxygen. This water normally serves as a source of demineralized water for initial filling of the Reactor Coolant Systems (RCSs) including the pressurizer relief tanks, the boric acid tanks, the Refueling Water Storage Tanks, the CCWSs, the Turbine Plant Cooling Water (TPCW) Systems, and the spent fuel pools. Demineralized water is also transferred to condenser hot wells, Condensate Storage Tanks, and Reactor Makeup Water Storage Tanks (RMWSTs) as required by their respective levels. The demineralized water is deaerated in a vacuum degasifier in the water treatment plant to remove dissolved oxygen before it is transferred to the Condensate Storage Tanks and the RMWSTs. The RMWST has a nitrogen sparger installed inside the tank. Nitrogen from the plant bulk low pressure nitrogen gas system is injected into the tank to reduce and maintain low dissolved oxygen levels.

Demineralized water is also provided via a transfer pump to the containment pre-access filtration units charcoal deluge system, and to the fire protection standpipe for the containment hose stations. For a detailed description of the Fire Protection System refer to Section 9.5.1.

Demineralized water is also required to perform turbine generator primary flow rate testing during refueling outages. The source of the demineralized water is provided at the inlet line to the turbine plant cooling water head tank.

Demineralized water is supplied to some eyewash stations and showers which are located upstream of any connections to radioactive portions of the system (e.g. Reactor Makeup Water) or connections to radioactive components (e.g. Refueling Water Storage Tanks) such that they are adequately protected from inadvertent radiation contamination. Either aerated or deaerated water may be used for this purpose. Deaerated water becomes aerated upon discharge from the spray nozzles and is acceptable for this use. Water downstream of the Reactor Makeup Water Storage Tank is not used for eyewash stations or showers.

The RMWST serves as a source of reactor makeup water and may contain trace amounts of tritium if it has been recycled from the RCS. Demineralized deaerated water is provided to the RCS and the auxiliary equipment, where the presence of tritium is not objectionable. These include supply to evaporators, gas strippers, pumps, demineralizer tanks, and pipelines for cleaning and flushing operations, and to the RCS as a diluent. The water chemistry for the reactor makeup water is listed in Table 9.2-10.

9.2.3.2 System Description

The Water Treatment System (which includes surface water pretreatment) is used to produce demineralized water, of quality per Table 9.2-9, through three new trains with a total capacity of 420 gpm, which is fed directly to the DWST. Figure 9.2-4 shows the paths used for treating lake water or well water in the Water Treatment System. The mixed-bed effluent flows directly to the DWST.

Sulfuric acid and sodium hydroxide are used to regenerate the Water Treatment System demineralizers. The spent regenerants are pumped to the Wastewater Management System for treatment prior to discharge, as discussed in Section 9.2.8.2.

Water from the DWST is used as a source for initial filling of the RMWST and Reactor Makeup Water System.

The RMWST, with makeup from the DWST if required, accommodates the makeup requirements resulting from a cold shutdown, followed by startup from cold conditions. A minimum quantity of water must be available during normal plant operation to achieve and maintain a safe, cold shutdown.

The RMWST is also required to provide a seismic category I supply of makeup water for the Chemical and Volume Control System, the Safety Chilled Water System, the Component Cooling Water System, and the Spent Fuel Pool Cooling System.

In the event of a loss of water from the spent fuel pools to the minimum suction elevation, there is normally sufficient volume in each RMWST to provide makeup to restore and maintain levels until cooling is restored.

As a minimum, one reactor makeup water pump per unit is normally operating.

If the system load demand is low, the pump recirculation insures that sufficient flow of tank water passes through the outdoor piping to prevent icing during severe outdoor temperatures.

The Demineralized and Reactor Makeup Water System is shown schematically in Figure 9.2-5.

The original system design intent was for Comanche Peak to be a “no release” plant, thereby recycling and reusing all effluent streams. For ALARA reasons, this philosophy has changed and the intent now is to prevent any tritiated or otherwise contaminated effluent from reaching the Reactor Makeup Water Storage Tanks. For this reason the isolation valves between the WP Waste Evaporator Condensate Pumps and the DD/RMW system are presently administratively controlled in a closed position. Also, the flowpath from the recycle evaporator package to the RMWST has been removed. In any case, previously designed processing of effluents from the

waste evaporator is possible by simply opening these valves and aligning other valves as appropriate.

9.2.3.3 Safety Evaluation

The Demineralized and Reactor Makeup Water System has sufficient storage capacity to supply makeup water to the system should the Water Treatment System be out of service for short periods of time. Two 50% capacity pumps and two trains of ion exchangers are provided to ensure the reliability of the Water Treatment System.

The Water Treatment System does not contain, treat, or produce any radioactive material in its operation. Therefore, any waste produced by this system can be deposited directly into the Low Volume Waste Treatment Facility.

The RMWST, reactor makeup water pumps, associated valves and piping, and all connections to the boric acid blenders, the safety chilled water surge tanks, the component cooling water surge tanks, and the spent fuel pools are designated seismic Category I and Safety Class 3. Containment penetrations are described in Section 6.2.4. Components of the Demineralized and Reactor Makeup Water System are compatible with all chemicals used throughout the system. Stainless steel piping is used throughout the system as a deterrent against corrosion in the transfer of corrosive fluids.

The RMWST is designed to withstand the effects of natural phenomena such as safe shutdown earthquake (SSE), probable maximum flood (PMF), and tornado missile. All components located in the Containment, Safeguards, Auxiliary, Fuel and Electrical, and Control buildings are located inside seismic Category I structures, which are also designed to withstand tornado effects. These seismic Category I structures are capable of withstanding adverse environmental conditions such as postulated earthquakes, tornadoes, and tornado missiles. (For details on seismic Category I structures, see Section 3.2.1, 3.3, and 3.5.)

Any failure of non-safety-class-design equipment associated with the Demineralized and Reactor Makeup Water System will not cause any failure of safety-related systems or components.

Flow restriction orifices are provided in each Unit's Reactor Makeup Water supply line to the Chemical and Volume Control System. These orifices are designed to limit the rate at which unborated water can be added to the RCS. For further information see section 15.4.6.

9.2.3.4 Testing and Inspection Requirements

The equipment in the Demineralized and Reactor Makeup Water System is initially inspected and tested to insure system integrity and completeness. The inspection includes the following:

1. Pumps and Motor Drives

   Each pump is started and runs for sufficient time to insure its proper operation. Maintenance is provided as required.

2. Valves

Each valve is operated through its complete range to ensure that it is in normal operating condition.

3. Safety Class Isolation Valves

   An electrical signal is transmitted to ensure that isolation valving systems (safety-related) are operating properly.

4. Instruments and Annunciators

   Operation is checked periodically for proper operation and accuracy.

9.2.3.5 Instrumentation Requirements

Control for the Demineralized and Reactor Makeup Water System is monitored from the main control panel or from local panels: the potable and demineralized water panel (PDP) or the water treatment panel (WTP), which are located adjacent to each other.

The Demineralized and Reactor Makeup Water System is controlled from a local PDP. System operation is designed to be fully automatic, but with provision for manual control. The final water quality of the Water Treatment System is continuously monitored for conductivity. High conductivity is alarmed locally. Deviations from specified water quality trip the demineralizers to prevent poor quality water from entering the DWST. Regeneration is manually initiated and then proceeds automatically or manually until completion.

Level indicators are provided in both the local panel and the Control Room for the DWST (PDP), RMWST (PDP), and vacuum deaerator (WTP) to provide level indications. The vacuum deaerator provides local level indication only. For the RMWST, temperature measurement is also provided with an indication on the main control panel. Vacuum deaerator pressure is also monitored and indicated locally. Pump discharge pressure is monitored and indicated locally for all pumps.

Flow measurements are provided for the main demineralized water supply line and the discharge line of the deaerated water transfer pumps. Signals from flow transmitters are indicated on the local panel with total flow indication on the main control panel.

9.2.4 POTABLE AND SANITARY WATER SYSTEM

9.2.4.1 Design Bases

The station potable and sanitary water system is designed to provide the following:

1. Water for toilets, sinks, showers, and drinking purposes in all permanent personnel areas of the plant site, as required.

2. Water for emergency eyewash and showers, as required.

3. Water to fire protection hoses for various on-site buildings.

4. Water to fill and to provide normal makeup to the Fire Protection Storage Tanks.

9.2.4.2 System Description

The distribution system that provides potable water to the plant and associated support structures and buildings for both Unit 1 and 2 is supplied by the Somervell County Water District (SCWD) public water system.

The potable water is chlorinated to conform with the Texas Department of Health regulations.

Domestic wastes generated on-site are transferred to and treated in, the 90,000-gpd domestic waste treatment facility. Domestic waste treatment effluent is treated with high intensity ultraviolet light for disinfection and odor reduction and is discharged to the Squaw Creek Reservoir in accordance with regulatory agency permitted effluent limits. The treatment facility sludge is dewatered and bagged for appropriate disposal. (See Figure 9.2-7)

9.2.4.3 Safety Evaluation

The potable and sanitary water system is designed without interconnection with, and is physically separated from, any radioactive sources, thus precluding the possibility of radioactive contamination. It is completely divorced from the laundry and hot shower portion of the Liquid Waste Processing System described in Section 11.2. Wastes produced by the potable and sanitary water system contain no radioactive materials and can therefore be safely treated in the domestic waste treatment facility.

The maximum radioactive release from the plant is below the limits established by 10 CFR Part 20, Appendix B. Contamination of ground water for normal operation is not considered significant; the concentration is well within 10 CFR Part 20, Appendix B limits. For an evaluation of the effects of accidental releases of liquid radioactive material, see Sections 2.4.12 and 2.4.13.3. Monitoring and safeguard requirements are discussed in Section 2.4.13.4.

Because the system is common to both units and is independent of their operation, a shutdown of either or both units does not affect the supply of potable water. Additionally, potable water can be trucked to the site and distributed in portable containers.

In case of water contamination (radiological or otherwise) or an accident where the piping is forced out of service, potable water can be trucked to the site and distributed in portable containers.

9.2.4.4 Tests and Inspection

1. Sampling of Potable Water

   Water samples are taken at various locations throughout the Potable Water System. Samples are collected and tested in accordance with regulatory agency standards.

9.2.5 ULTIMATE HEAT SINK

9.2.5.1 Design Bases

The function of the ultimate heat sink is to dissipate heat rejected from the Station Service Water System during postaccident shutdown and normal cooldown conditions. It has been sized to

provide adequate cooling capacity for the CPNPP as required by NRC Regulatory Guide 1.27, Rev. 2.

The ultimate heat sink complex of the CPNPP consists of the Safe Shutdown Impoundment (SSI) (the body of cooling water) and the SSI dam, as shown in Section 2.4 and on Figure 2.4-1.

There are two possible conditions of operation for the SSI: 1) the simultaneous shutdown and cooldown of both units or 2) the shutdown and cooldown of one unit concurrent with the dissipation of post-design-basis-accident (DBA) rejected heat from the other unit. As required by NRC Regulatory Guide 1.27, the SSI has been sized for the most severe of these two alternatives, which, in the case of the CPNPP, is the first condition. In addition, it is assumed that the Squaw Creek Reservoir (SCR) is lost coincident with the DBA and that no makeup water is added to the SSI for the 30-day period of evaluation following the DBA. Therefore, all of the 30-day postaccident cooling requirements are satisfied by the amount of water contained in the SSI with the initial water level at elevation 770 ft 0 in. followed by loss to elevation 769 ft 6 in., which is the limit of entrainment set by the invert of the equalization channel between the SSI and the SCR. (See Section 2.4.)

The ultimate heat sink design is in accordance with the following GDC and NRC Regulatory Guides:

1. GDC 2 [1], 4 [2], 5 [3], 44 [5], 45 [6], and 46 [7].

2. NRC Regulatory Guides 1.27 [11] and 1.29 [12].

Specifically, the ultimate heat sink complex can withstand the effects of the SSE, tornado, or flooding effects of the probable maximum flood (PMF) without loss of safety function.

9.2.5.2 System Description

The SSI is an enclosed body of water formed from a cove of the SCR and is retained by a seismic Category I dam, designed and constructed to withstand the most severe postulated natural phenomena, as described in Chapter 2. The SSI dam separates the SSI from the SCR. A complete description of the SSI dam is given in Section 2.4.

An equalization channel allows each body of water to adjust to a common level above the minimum water level. In the event the SCR dam fails, this channel limits the low water level in the SSI to 769 ft 6 in., at which point the volume of water contained is approximately 284 acre-feet, allowing for 40 years of sedimentation. During normal operation, the water level is at elevation 775 ft with a volume of 467 acre-feet contained. A non-intrusive, floating raft system has been placed near the south end of the equalization channel. The raft system has been designed to allow water to freely pass under the raft for all water levels without blocking flow during severe natural phenomena such as a postulated seismic or PMF event.

A bleed line is provided from the Circulating Water System upstream of the main condensers, at the CWS intake structure, to the Service Water Intake Structure pump suction pit. The circulating water bleed-off was designed to ensure continuous makeup to the SSI and provide constant outflow (blowdown) to prevent excessive dissolved solids concentration in the SSI. The bleed line is not in operation due to concerns about the piping's internal liner. Make-up is not required

because of the equalization channel described above. Dissolved solids are reduced by normal rainfall and runoff.

9.2.5.3 Safety Evaluation

The heat rejection capabilities of the SSI are a function of meteorological conditions, the volume and surface area of the SSI and the location of structures. All three of these features can be represented by using a numerical, three-dimensional, time-varying hydrodynamic and transport model to simulate the performance of the SSI.

An analysis of 39 years of offsite meteorological data showed that August 31, 1990 was the most severe period for ultimate heat sink performance, both for the 24 hour transient analysis and for peak SSI intake temperatures. These data are shown in Table 2.3-7B. Similarly, the period June 25, 1980 to July 25, 1980 was shown to be the most severe period of evaporative water loss. The meteorological data for this period is shown in Table 2.3-7C. See Section 2.3.1.2.10 for details.

The hydrodynamic and transport model numerically solves the three-dimensional momentum, mass and constituent balance equations. The model requires a finite difference representation of the SSI bathymetry, locations and dimensions of the discharge and intake structures and of the equalization channel connecting the SSI with SCR. These data were generated from topographic and design drawings.

Heat rates of the cooling water discharged onto the SSI are based on two safety trains in operation per unit, and are developed from the nuclear steam supply system (NSSS) manufacturer's functional requirements, design criteria for residual decay heat removal, and from balance-of-plant (BOP) heat load requirements. In generating the spent fuel pool heat load, it was assumed that the DBA occurs in one unit during normal two unit operation during summer conditions. The spent fuel pool heat loads for summer conditions are described in Section 9.1.3.1.

The decay heat rate, however, is based on BTP ASB 9-2. This results in a high heat input which reflects a conservative spent fuel pool heat loading in conjunction with a conservative containment spray and RHR heat loading.

1. Compliance with NRC Regulatory Guide 1.27

   The intent of NRC Regulatory Guide 1.27 is met by SSI. The thermal ability of the SSI to act as an ultimate heat sink for 30 days is covered in Item 3, Thermal Performance Evaluation. As a single source ultimate heat sink serving two units, the SSI is able to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, taken individually, without loss of capability to perform its safety functions. The natural phenomena and their magnitude are selected in accordance with their probability of occurrence, and designs are based upon the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in historical data. Such phenomena and design criteria are discussed in Sections 2.4, 3.3, and 3.4.

2. Hydraulic Performance

The SSI is supplied with a bleed flow from the Circulating Water System that may be operated to supply a continuous flow of water. This flow is directed into the service water pump intake structure and can be used if required as a blowdown flow from the SSI to the SCR through the equalization channel. If, as a result of an earthquake, the SCR dam fails, the equalization channel invert maintains the water level in the SSI at elevation 769 ft 6 in. A floating raft system in the equalization channel is non-intrusive relative to flow through the channel and does not adversely affect the performance of the function of the SSI. Surface area and volume in the SSI as a function of elevation is discussed in Section 2.4.

Hydraulic short circuiting is prevented by the physical separation of the intake structure and discharge piping outfall and the orientation of the discharge. The intake and discharge points are over 1800 ft apart, and the exit velocity of the discharge water carries it upstream initially, away from the area of the intake structure, allowing extra time for the transfer of heat to the atmosphere.

The service water system effluent flows through an open channel type discharge canal prior to entering the SSI. Cooling occurs throughout the approximately 1300 ft. length of the discharge canal before the service water effluent mixes with the SSI bulk fluid. The discharge canal outlet enters the SSI in a direction away from the service water intake structure. Mixing and dispersion of the effluent occurs when the SSI fluid reverses direction in order to pass to vicinity of the intake structure. The minimum shore line distance between the discharge canal and intake structure is 1500 ft. Further cooling of the mixed fluid occurs in route. Because the intake is submerged and the discharge is on the surface, there is vertical as well as horizontal separation. See Figure 1.2-46 for the intake structure.

3. Thermal Performance Evaluation

To simulate the operation of the SSI preceding and following LOCA, the time varying meteorological data must be aligned with the heat load data such that the peak SSI intake temperature due to the imposed heat load occurs at the same time that the natural temperature peak occurs. An additional consideration is that the simulation be run for a sufficiently long time so that the SSI temperature reaches stationary state with respect to the pre-LOCA heat load. Both these issues were addressed with sensitivity simulations. To determine the initialization time, steady meteorological data were used with a steady heat load to determine that the SSI temperatures reached steady values after 37 days. To determine the start of LOCA relative the single worst day for atmospheric cooling, the time-varying heat loads were run with steady meteorological data. These tests showed that the time to peak intake temperature due to time varying heat loads and plant pumping was 10 days. To further refine this number, additional sensitivity tests were made in which the time to peak was varying under both time-varying meteorological and heat load conditions. The resulting time to peak was found to be seven days.

The LOCA intake temperature reaches a maximum on the evening of the seventh day after a LOCA at less than 116°F (two train).

A 24-hour transient analysis for a two train ESF LOCA case was conducted with the model to determine how quickly intake temperatures would rise immediately after a

LOCA. The lag between discharge and intake temperature rises was determined to be more than 5 hours.

Component cooling water temperature is a function of heat load on the system and the service water temperature. Component cooling water supply temperature in the DBA unit peaks at approximately 135°F, upon initiation of containment spray recirculation, with service water temperature at 102°F. The maximum component cooling water temperature of the shutdown unit, with only one heat exchanger operable, is limited to 122°F. This temperature occurs upon initiation of RHR, several hours after the DBA occurred in the other unit. The preceding maxima of 135°F and 122°F ensure that the system will perform satisfactorily in mitigating the event in the DBA unit concurrent with orderly cooldown of the shutdown unit.

During the postulated 100-year drought conditions and after 40 years of sedimentation, the SSI is determined to have 284-acre feet of water. Although the design basis (RG1.27) requirement is for a 30-day cooling supply, the consumption analysis was based on a 39-day post-accident shutdown period. The maximum consumption of SSI water during the 39-day postaccident shutdown cooldown period amounts to approximately 94 acre-feet, resulting in a decrease in surface elevation of 3.9 ft., allowing adequate margin for post-30-day operation without exceeding the service water pump submergence requirements. Refer to Sections 2.4.11.5 and 2.4.11.6 for further discussion of the heat sink dependability requirements.

A loss of offsite power for both units was evaluated using the same methodology as for LOCA described above. The intake temperature for this scenario reaches a maximum on the evening of the seventh day at 117°F. The maximum water consumption at 30 days amounts to approximately 78.4 acre-feet, resulting in a surface elevation of 766 ft. 4 in. The maximum water consumption at 39 days is approximately 99.4 acre-feet, resulting in a surface elevation of 765 ft. 4 in.

Consumption of SSI water by the Auxiliary Feedwater System for supply to the steam generators, if a beyond the design basis failure of one Condensate Storage Tank is assumed, amounts to only approximately 0.67 acre-feet, based on a 60 gal of water per thermal MW rating of the steam supply system.

The time variation of decay power, based on the ANS 5.1 fission product curve, is corrected for a finite operating time and includes the ANS uncertainty factors. The related curve has been used to develop the values of decay heat rates shown in Table 9.2-13.

The heat rejection capabilities of the SSI are a function of the volume/surface area relationship of this body of water. The SSI is sufficiently sized to accept plant-rejected heat under the most severe conditions specified in NRC Regulatory Guide 1.27, Rev. 2.

9.2.6 CONDENSATE STORAGE FACILITIES

9.2.6.1 Design Bases

The condensate storage facilities (Figure 10.4-11) provide makeup and surge capacity for secondary system inventory changes caused by different operational conditions, thermal effects, and the draining and recharging of any part of the system. It also provides sufficient water

storage (see Section 10.4.7.2 for additional details) for emergency decay heat removal by the Auxiliary Feedwater System in the event the Condensate and Feedwater System is inoperable. This amount of feed quality water is required to hold one unit for four hours at hot standby and then cool down the RCS at 50°F per hour for five hours, at which point the RHR System can take over [5].

The facilities are designed to seismic Category I, nuclear Safety Class 3 requirements [10],[12],[15]. The design conditions and criteria are those defined in Chapter 3. In addition, these facilities meet the requirements of the applicable codes listed in Section 3.8.4.

9.2.6.2 System Description

The condensate storage facility consists of one reinforced concrete, stainless steel-lined tank per unit, located in the plant yard adjacent to the respective diesel generator building (see Table 10.4-8 for additional details). The tank contains deaerated water of feedwater quality. A floating diaphragm arrangement prevents air leakage into the tank. The tank's stainless steel liner will not be attacked by deaerated demineralized water. The tank contains water which is used as storage for the Condensate System (see Section 10.4.7.2 for additional details). The reserve auxiliary feedwater (see Section 10.4.7.2 for additional details) cannot be drained by the non-nuclear-safety-related systems because of the elevation of the outlet nozzles. The tank receives makeup water from the Demineralized Water System.

Inadvertent drainage of the tank is prevented by normally closed double valves. In addition, automatically operated valves isolate all other uses of the Condensate Storage Tank when auxiliary feed operation is required, thereby maximizing the volume of water available for safety-related purposes. Refer to Section 10.4.9 and Figure 10.4-11 for the Condensate Storage Tank design parameters.

9.2.6.3 Safety Evaluation

The Condensate Storage Tank is an ANS Safety Class 3 structure. Since this facility is designed to seismic Category I requirements (Section 3.7), failure is not postulated, and there are no structural and resultant environmental effects. Adequate ability of the tank to withstand earthquakes is ensured by designing the tank in accordance with the requirements for seismic Category I structures. The tank is also designed to withstand tornadoes and missile impact (Sections 3.3 and 3.5) [1],[2],[14]. Adequate protection against corrosion is ensured by a corrosion-resistant, ASTM A 240, chromium nickel stainless steel type 304L, 3/16-in.-thick liner.

Figure 10.4-11 shows the location of the CST vents. Each vent line alone provides the required venting capacity for normal operation and design basis accidents. The vents are located above the tank overflow level and their blocking by freezing water is not credible. The CST vents are classified as non-Nuclear Safety related and, in accordance with ANSI N18.2a, are designed to break away if struck by a tornado missile.

It is not anticipated that any significant radioactive material is present in the system.

Piping for the Condensate Storage Tank is located in the pipe tunnel and tank room. These areas are open to the Safeguards Building and are provided with thermostatically controlled area heaters. Freezing or icing under the most severe outdoor temperatures is not anticipated.

Environmental design conditions for seismic Category I self-actuated valves are described in Section 3.11.

9.2.6.4 Tests and Inspections

The Condensate Storage Tanks are tested during the preoperational test program for both the Condensate System and the Auxiliary Feedwater System. Periodic visual inspections are performed to ensure integrity of the tank [6],[7].

9.2.6.5 Instrumentation Requirements

Filling of the Condensate Storage Tank may be performed manually by the operator aligning demineralized water, or automatically by associated level instrumentation when the demineralized water source is continuously aligned. In the automatic makeup mode, Condensate Storage Tank makeup is supplied automatically whenever the tank level is below set point level. The demineralized water supply valve is opened to supply makeup until a high tank level is reached and the makeup valve is then closed. Makeup water can be supplied manually from a control board switch. Tank level is indicated locally and remotely in the Control Room. High-high, Low, and low-low tank level alarms are provided [4]. Redundant level transmitters are used. One level transmitter is used for makeup control and indications; the other transmitter is used for level monitoring and alarms.

The Condensate Storage Tank is isolated from its non-safety-related users by automatically closing motor-operated valves when the motor-driven or turbine-driven auxiliary feedwater pumps start. The condensate make-up and reject line is isolated by automatically closing motor operated valves when the condensate storage tank reaches a HI-HI level. Valves can be opened manually with control board switches with three positions, close, auto, and open, with spring return to auto.

9.2.7 SURFACE WATER PRE-TREATMENT SYSTEM

9.2.7.1 Design Bases

Surface Water Pre-Treatment System is designed to mitigate the possible consequences of ground water pumping at CPNPP by providing all water required for plant make-up.

9.2.7.2 System Description

Figure 9.2-4A shows a schematic flow diagram of Surface Water Pre-Treatment System. This treatment facility takes raw water from Squaw Creek Reservoir at the circulating water intake structure. Water is supplied from supply pumps installed at the intake structure. Removal of suspended matter and chlorination is accomplished by a pre-treatment (micro filtration) system. A reverse osmosis system reduces the dissolved solids. The micro filtration units produce a maximum of 800 gpm of filtered water. Storage tanks for filtered water and reverse osmosis product water provide system reliability and operation flexibility.

The supply pumps supply approximately 240 gpm to 800 gpm to the micro filtration system. The pre-treatment facility provides for combined filtration, collection, and discharge of suspended matter. A chemical feed system is provided for the addition of chemicals to aid in chlorination,

which is automatically controlled to adjust varying flow rates. Feed concentration of chemicals is adjusted periodically to meet the changing inlet water conditions.

Flow from the micro filtration system supplies the Filtered Water Storage Tank to control changes in flow rates.

9.2.7.3 Safety Evaluation

The Surface Water Pre-Treatment System is designed without interconnection with, and is physically separated from, any radioactive sources, thus precluding the possibility of radioactive contamination. It is completely divorced from the laundry and hot shower portions of the Liquid Waste Processing System described in Section 11.2. Wastes produced by the blowdown of the water treatment equipment contains no radioactive materials and can therefore be pumped to the Wastewater Management System, for treatment prior to discharge.

Because the system is common to both units and is independent of their operation, a shutdown of either or both units does not affect the supply of water.

9.2.7.4 Inspection and Testing Requirements

As the components of this system are in either continuous or intermittent use during normal plant operation, no additional periodic tests are required, although periodic visual inspections and preventive maintenance are conducted as necessary. All components are accessible for periodic inspection.

9.2.7.5 Instrumentation Requirements

The Surface Water Pre-Treatment System is designed for semiautomatic operation. The system is provided with audio and visual annunciation to alert the operator of abnormal conditions.

9.2.7.6 Water Treatment System

The Water Treatment System will normally produce 420 gpm of processed water for plant operational use. The primary equipment required to support this production rate includes two micro filtration units, Multi Media Filters (normally all three Multi Media Filters are in service), First Pass Reverse Osmosis Banks, the Forced Draft Decarbonator, Second Pass Reverse Osmosis Banks, the Vacuum Degasifier and Mixed Bed Demineralizers.

A flow control valve to the micro filtration units modulates flow into the unit. The chemical feeds provided for the micro filtration units include Sodium Hypochlorite, Caustic and Acid. Each chemical has at least one storage tank and a metering pump.

The micro filtration units are a membrane based filtration system which removes suspended solids and achieves water quality requirements in a one-step filtration process.

The Filtered Water Storage Tank receives the water, once it has been filtered. The Filtered Water Forwarding Pumps transfer water from the Filtered Water Storage Tank to the Multi Media Filters. Filtered water also serves as the backwash media for the Multi Media Filters.

Water from the Multi Media Filters is further processed with bisulfite, Cartridge Filters, and antiscalant injection. The bisulfite, sulfuric acid and antiscalant have at least one storage tank and two metering pumps each. There are three Cartridge Filter units.

Water is then transferred via the First Pass Reverse Osmosis Feed Pumps to the three First Pass Reverse Osmosis Banks. The First Pass Reverse Osmosis Banks are designed to reduce the total dissolved solid content of the water.

From the First Pass R.O. product, water is sent through a Forced Draft Decarbonator, where carbon dioxide concentration in the water is reduced.

After decarbonation, the Second Pass Reverse Osmosis Feed Pumps will transfer the processed, decarbonated water to the Second Pass Reverse Osmosis Feed Banks. The Second Pass R.O. Feed Banks will further reduce the effluents total dissolved solids content. Following this treatment, water is transferred to the R.O. Product Water Storage Tank.

R.O. Product Water Forwarding Pumps transfer water from the R.O. Product Water Storage Tank to the Vacuum Degasifier. The Vacuum Degasifier will reduce the dissolved oxygen content of the effluent. Effluent from the Vacuum Degasifier then enters the Mixed Bed Demineralizer/Polisher Units.

The Mixed Bed Units provide the final demineralizing/polishing treatment of the water which is then transferred to the Demineralized Water Storage Tank for use in the plant.

9.2.8 WASTE MANAGEMENT SYSTEM

9.2.8.1 Design Bases

The Waste Management System (WMS) provides wastes collection, retention, conventional pollutant treatment, and discharge of normally non-radioactive wastewaters in accordance with the regulatory requirements of the Texas Commission on Environmental Quality (TCEQ) and the Texas Pollutant Discharge Elimination System (TPDES) permits. The WMS is shown on Figure 9.2-15 and 9.2-16.

The WMS collects wastewaters for processing in either the Low Volume Waste (LVW) or Co-Current Waste (COW) treatment facilities. The WMS also provides for low level radioactive contaminated Condensate Polisher resins to be transferred to the LVW treatment facilities for retention and subsequent proper disposal in accordance with regulatory requirements.

9.2.8.2 System Description

9.2.8.2.1 Low Volume Waste Treatment Facilities

The Low Volume Waste (LVW) treatment facilities provide collection, treatment and discharge of normally non-radioactive wastewaters. The LVW facilities also provide for settling and retention for normally non-radioactive Condensate Polisher resins and the SWPS clarifier sludge. As shown in Figures 9.2-15 and 9.2-16, wastes from several sources are normally processed through the LVW treatment facilities. These sources are:

1. Turbine Building sumps

2. Auxiliary Building Sump #11

3. Diesel Generator Room sump (2 per unit)

4. Component Cooling Water Drain Tanks (1 per Unit)

5. Surface Water Pretreatment System backwash

6. Surface Water Pretreatment System Reverse Osmosis reject and rinse water

7. Condensate Polisher Backwash Waste Cycle wastes

8. Chemical sump wastes from DWMS regeneration and leakage

9. Condensate Polishing Unit Phase Separators Solids

10. Surface Water Pretreatment System (SWPS) Sludge blowdown

The LVW Settling Ponds consist of two 1,750,000 gallon capacity lined settling ponds and a 6,700,000 gallon capacity emergency settling pond. The 1,750,000 gallon settling ponds are sized for average water flow of approximately 455,000 gallons per day to be turned over in accordance with the ODCM limits and waste water discharge permit (batch or continuous flow). These ponds are lined with a double synthetic liner with a leachate collection system.

The non-radioactive low volume wastes from secondary support systems include equipment, floor, laboratory, and sample drains; water treatment wastes from demineralizer regeneration, reverse osmosis (RO) systems operation, condensate polisher system and other miscellaneous water treatment blowdown and backwash operations; periodic drainage and flushing of various system components: and intermittent boiler blowdown from the plant's auxiliary boiler. These low volume wastes and the previously monitored wastes discharged from the Metal Cleaning Waste Outfall are routed to a Low Volume Waste Management System (WMS) and discharged via the Low Volume Waste Outfall. The WMS consists of the following components: surge basin, API oil separator, clarifier blowdown and condensate polisher decant basins, three separate but interconnected low volume waste retention ponds with double synthetic liners and leachate collection systems, and a synthetic lined concrete equalization basin. The complete WMS provides for oil and grease removal, suspended solids reduction through settling or further treatment, if necessary, and pH control through addition of acid or caustic, if required, to meet TCEQ permit limitations.

The treated non-radioactive low volume waste are normally commingled with the wastes which are potentially low level radioactive and are discharged via the Low Volume Waste Outfall to the condenser cooling water; the commingled waste stream is sampled prior to mixing with cooling water for compliance with the appropriate effluent limitations.

The LVW retention pond effluents are normally within Texas Commission on Environmental Quality (TCEQ) and TPDES permit limits for discharge without additional treatment. Monitoring to verify compliance with these permit limits is performed.

All LVW discharges flow through the Discharge Flume for flow rate monitoring.

A Metal Cleaning Waste (MCW) Pond has been dedicated for the storage and treatment of metal cleaning wastes. Metal cleaning wastes are generated by the chemical cleaning of plant equipment and are evaluated prior to their collection and retention in the pond. Unsuitable wastes area treated and/or disposed by alternate methods; e.g., off-site contract disposal.

Wastewater from the pond, as well as treatable metal cleaning wastes which might be collected in temporary facilities, is treated for metals and pH adjustment prior to discharge via the Metal Cleaning Waste Outfall to the Low Volume Waste retention ponds associated with the Low Volume Waste Outfall. The treated wastes so discharged will be commingled as a previously monitored effluent with the non-radioactive low volume wastes prior to discharge via the Low Volume Waste Outfall.

Chemical cleaning wastes or other metal cleaning wastes can be trucked to the 1,000,000 gallon MCW Pond. The wastes from this pond can be treated for release through the Metal Cleaning Waste Outfall or retained to allow evaporation or further processing. The MCW Pond is sized to accommodate six (6) volumes of the four (4) steam generators (per unit) and associated piping. This pond is lined with a double synthetic liner with a leachate collection system.

Several components of the old evaporation pond wastewater treatment system have been abandoned in place. These components include but are not limited to: batch neutralization tank, LVW oil/water separator skid, soda ash mix tank, sulfuric acid tanks, LVW transfer pumps, clear well transfer pumps and other associated electrical and mechanical equipment. These components have been deleted from the FSAR figures and will be removed from the plant and salvaged.

9.2.8.2.2 Co-Current Waste Treatment Facilities

The Co-Current Waste (COW) treatment facilities provide collection, batch hold-up, conventional pollutant treatment and discharge of low level radioactive waste in accordance with environmental and NRC regulatory requirements. Except for powdex resins, potentially contaminated low level radioactive waste normally transferred to the LVW treatment facilities are transferred to the COW treatment facilities when radioactive contamination is detected in excess of the limits specified in the Radiological Effluent Controls Program required by Technical Specifications. As shown in Figure 9.2-15, potentially low level radioactive waste collected in the following areas can be diverted to the COW treatment facilities:

1. Turbine Building sumps

2. Auxiliary Building Sump #11

3. Diesel Generator Room sump (2 per unit)

4. Component Cooling Water Drain Tank (1 per unit)

5. Condensate Polisher Backwash Waste Cycle wastes

6. Auxiliary Building Sump #3

Diverted wastes are collected in the Waste Water Hold-up Tanks to facilitate batching. The batched wastes are thoroughly mixed by mechanical mixing and/or placing the hold-up tank

contents in a recirculation mode. While recirculating, the batched wastes can be sampled and analyzed to quantify low level radioactive contents.

The environmental regulatory requirements for pH, oil and grease, and total suspended solids in the discharged wastes are specified in Texas Commission on Environmental Quality (TCEQ)-TPDES permits.

Batched volumes requiring pH adjustment are manually neutralized. Following pH adjustment the batched wastes may be discharged through the environmental Low Volume Waste Outfall. If required, the wastes are treated for emulsified oil removal and/or suspended solids reduction prior to discharge through the Low Volume Waste Outfall. Alternately, the wastes may also be drained to Turbine Building Sump No. 4 and discharged to the Low Volume Waste retention pond if radioactivity concentrations are below levels specified in the Radiological Effluent Controls Program required by the Technical Specifications.

REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, Design Bases for Protection Against Natural Phenomena.

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, Environmental and Missile Design Bases.

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, Sharing of Structures, Systems, and Components.

4. 10 CFR Part 50, Appendix A, General Design Criterion 19, Control Room.

5. 10 CFR Part 50, Appendix A, General Design Criterion 44, Cooling Water.

6. 10 CFR Part 50, Appendix A, General Design Criterion 45, Inspection of Cooling Water System.

7. 10 CFR Part 50, Appendix A, General Design Criterion 46, Testing of Cooling Water System.

8. 10 CFR Part 50, Appendix A, General Design Criterion 56, Primary Containment Isolation.

9. 10 CFR Part 50, Appendix A, General Design Criterion 57, Closed System Isolation Valves.

10. NRC Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive- Waste-Containing Components of Nuclear Power Plants, Revision 3, February 1976, U.S. Nuclear Regulatory Commission.

11. NRC Regulatory Guide 1.27, Ultimate Heat Sink for Nuclear Power Plants, Revision 1, March 1974, U.S. Nuclear Regulatory Commission.

12. NRC Regulatory Guide 1.29, Seismic Design Classification, Revision 2, February 1976, U.S. Nuclear Regulatory Commission.

13. NRC Regulatory Guide 1.48, Design Limits and Loading Combinations for Seismic Category I Fluid System Components, May 1973, U.S. Nuclear Regulatory Commission.

14. Branch Technical Position APCSB 3-1, Protection Against Postulated Piping Failure in Fluid Systems Outside the Containment.

15. ANSI N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, 1973.

16. American Nuclear Society Draft Standard ANS 5.1, Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors, October 1973.

17. Generic Letter 91-13, “Request for Information Related to the Resolution of Generic Issue 130, Essential Service Water System Failures at Multi-Unit Sites, Pursuant to 10 CFR 50.54(f),” dated September 19, 1991.

18. TXX-92410, License Amendment Request 92-002, Combined Unit 1 and 2 Technical Specifications, from W. J. Cahill, Jr., to U. S. Nuclear Regulatory Commission, dated August 31, 1992.

19. Generic Letter 89-13, “Service Water Problems Affecting Safety Related Equipment”, dated July 18, 1989, and Supplement 1, dated April 4, 1990.

TABLE 9.2-1
SINGLE-FAILURE ANALYSIS OF STATION SERVICE WATER SYSTEM

Component	Malfunction	Effect on System Safeguard Performance	Comments
Service water pump	stops pumping	No effect	Two 100-percent-capacity pumps are provided; one is required.
SSW pump discharge isolation valve	a. Fails to close b. Unwanted closure c. Fails to open	a. No effect b. No effect c. No effect	a. Pump shutdown and check valve prevent flow in either direction. b. Redundant train is provided. c. Redundant train is provided
Emergency diesel generator isolation valve	a. Fails to open b. Fails to close	a. No effect b. No effect	a. Two 100-percent-capacity trains are provided; one is required. b. Not required to close to accomplish a safety function.
Auxiliary feedwater isolation valve	a. Fails to open b. Fails to close	a. No effect b. No effect	a. Two 100-percent-capacity trains are provided; one is required. b. Required minimum flow and sufficient NPSH required are available to AF pumps.
Emergency diesel generator	Fails to start	No effect	Two 100-percent-capacity diesel generators are provided per unit.
Component cooling heat exchanger	Loss of flow	No effect	Two 100-percent-capacity diesel generators are provided per unit.

TABLE 9.2-2
COMPONENT COOLING WATER SYSTEM EQUIPMENT CHARACTERISTICS

(Sheet 1 of 2)

	Shell Side	Tube Side
1. Heat Exchanger		
Design flow, gpm	14,700	14,000
Design pressure drop, psi	12.4	5.3
Design pressure, psig	165	150
Design temperature, °F	225	225
Temperatures, °F		
Outlet	105.0	108.4
Inlet	114.5	98.4
Design heat transfer rate	70×10^6 BTU/hr.	
Tubes Material	ASME SB-111 Type 706	
2. Pumps		
Type	Centrifugal, horizontal	
	Design	Runout
Capacity gpm	14,700	16,400
Total Dynamic Head, ft	226	210
NPSH, ft		
Minimum required	30	36
Available (minimum)	60	60
Design pressure, psig	165	
Temperature, °F		
Normal	115	
Normal (max.)	180	
Design	225	
Material	Carbon steel	

TABLE 9.2-2
COMPONENT COOLING WATER SYSTEM EQUIPMENT CHARACTERISTICS

(Sheet 2 of 2)

3. Surge Tanks	
Outside diameter, ft	6.5
Length, ft	22
Volume, gal	4600
Design pressure, psig	10
Design temperature, °F	200
Material	Carbon steel

4. Piping

All CCWS piping is carbon steel material, ASME SA 106, Grade B, with welded joints and connections except at components where flanged connections are used to facilitate maintenance.

TABLE 9.2-3
COMPONENT COOLING WATER SYSTEM WATER CHEMISTRY

1. Acceptable Corrosion Inhibitors

   Any of the following corrosion inhibitors can be used:

   a. Sodium Molybdate

   b. Tolyltriazole

   c. Hydrazine

   d. Calgon Corporation's H-300 (Gluteraldehyde)

   e. Sodium Nitrite

   f. Borax

   g. Chromates

   h. Sodium Hydroxide

2. Chloride, ppb <150

3. Fluoride, ppb <150

TABLE 9.2-4
COMPONENT COOLING WATER SYSTEM IN RELATION WITH CODES AND STANDARDS

Plant Conditions	Condition I Normal Operation	Condition II Incidents of Moderate Frequency	Condition III Infrequent Incidents	Condition IV Limiting Faults
NRC Regulatory Guide 1.48 plant conditions	Normal	Upset	Emergency	Faulted
ASME Section III Component Conditions				
Pumps, valves and vessels	Normal	Normal	Normal	Normal
Piping	Normal	Upset	Emergency	Emergency
System Operation	Startup; standby; power range; refueling; cooldown; 4 hrs; cooldown, 20 hrs	Loss of outside power; reactor-turbine mismatch	Small ruptures	LOCA

TABLE 9.2-5
SINGLE FAILURE CRITERIA ANALYSIS OF COMPONENT COOLING WATER SYSTEM

(Sheet 1 of 4)

Component			Malfunction		Effect on Safeguards Systems	Comments
1.	Component cooling water pump		Stops pumps or fails to start		No effect	Redundant train is provided.
2.	Component cooling heat exchanger		Tube or shell rupture		No effect	Redundant train and automatic isolation is provided.
3.	Remotely Operated Stop Valves					
	a.	Containment isolation	1)	Fails to close	No effect	Backup provided by redundant valve or by closed system inside containment.
			2)	Unwanted closure	No effect	Closure of nonsafeguards valves does not affect safe shutdown.
	b.	Containment spray and RHR heat exchanger isolation	1)	Fails to open	No effect	Valve has a manual opeator; heat exchanger redundancy is provided.
			2)	Unwanted closure	No effect	Valve has a manual operator; heat exchanger redundancy is provided.
	c.	Non-safety loop isolation	1)	Fails to close	No effect	Valve redundancy is provided.
			2)	Unwanted closure	No effect	Valve has a manual operator.

TABLE 9.2-5
SINGLE FAILURE CRITERIA ANALYSIS OF COMPONENT COOLING WATER SYSTEM

(Sheet 2 of 4)

Component			Malfunction		Effect on Safeguards Systems	Comments
	d.	Safety loops	1)	Fails to close	No effect	Valve redundancy is provided.
			2)	Unwanted closure	No effect	Loop redundancy is provided
4.	Safety-related headers		Pipe rupture		No effect	Redundant headers and automatic isolation are provided.
5.	Emergency diesel generators		Fails to start		No effect	Redundant diesel generator is provided.
6.	Motor Control					
	a.	CCW Pump Motor	Fails to start due to:			
			1)	Power loss	No effect	Redundant and independent train provided for backup pump on second loop. ESFAS circuitry activates both pumps' motors.
			2)	Failure in control	No effect	
	b.	Safety class and non-safety class loop isolation valves.	Fail to close:			Redundant and independent train provided for backup valve. ESFAS activtes all loop isolation valves in both trains.
			1)	Power loss	No effect	
			2)	Failure in control circuitry	No effect	

TABLE 9.2-5
SINGLE FAILURE CRITERIA ANALYSIS OF COMPONENT COOLING WATER SYSTEM

(Sheet 3 of 4)

Component		Malfunction		Effect on Safeguards Systems	Comments
c.	Containment spray and RHR heat exchanger Isolation Valves	Fail to open			Redundant and independent train provided for isolation valve on backup safety class loop.
		1)	Power loss	No effect	
		2)	Failure in control circuitry	No effect	
d.	CCW system phase B Containment Isolation Valves	Fail to close			Redundant and independent train provided for backup valve. ESFAS activates valves on both trains.
		1)	Power loss	No effect	
		2)	Failure in control circuitry	No effect	
7.	Air Operated Valves				
a.	Phase “A” CCW system Containment Isolation Valves	1)	Loss of Instrument Air	No effect	Valves fail in safe position on loss of air.
		2)	Power loss	No effect	Valves fail in safe position on loss of power
b.	Safety Injection CCW system ventilation condenser isolation	1)	Loss of Instrument Air	No effect	Valves fail in safe position on loss of air.
		2)	Power loss	No effect	Valves fail in safe position on loss of power.
		3)	Fail to close (1 valve)	No effect	Two valves are provided to stop flow to the condensers.

TABLE 9.2-5
SINGLE FAILURE CRITERIA ANALYSIS OF COMPONENT COOLING WATER SYSTEM

(Sheet 4 of 4)

Component		Malfunction		Effect on Safeguards Systems	Comments
c.	Surge Tank Makeup Water valves	1)	Loss of air	No effect	Reactor makeup water valve fails in safe position.
		2)	Loss of power	No effect	Fill valves are provided with handwheels.
		3)	Reactor Makeup Valve fails to open	No effect	Same as loss of air. Valve will be forced open by operator.

TABLE 9.2-6
NOT USED

TABLE 9.2-7
NOT USED

TABLE 9.2-8
WELL WATER ANALYSIS

Constituent	Concentration (mg/1)
Calcium, as Ca	17
Magnesium, as Mg	8
Sodium, as Na	141
Carbonate, as CO3	0
Bicarbonate, as HCO3	378
Sulfate, SO^4	39
Chloride, as C1	22
Silica, as SiO2	12
Total ion, as Fe	0.05
Manganese, as Mn	0.02
Total dissolved solids	615
Phenolphthalein Alkalinity, as CaCO3	0
Total alkalinity, as CaCO3	310
Turbidity, units	1
Color, units	5
PH	8.55
Total Hardness, as CaCO3	77
Free carbon dioxide, as Co2	16

TABLE 9.2-9
DEMINERALIZED WATER ANALYSIS

Constituent	Concentration
Specific conductivity, micromhos/cm or microSiemens/cm	<0.1
Soluble silica as SiO2, ppm	<0.05
Suspended solids, ppm	<0.05
pH at 25°C	6.0 to 8.0
Chloride, ppm	Negligible
Fluoride, ppm	Negligible

TABLE 9.2-10
REACTOR MAKEUP WATER

Constituent	Concentration
Cation conductivity, micromhos/cm or microSiemens/cm at 25°C	≤ 1.0
PH	6.0 to 8.0
Oxygen, ppm[a]	≤ 0.10
Total Chloride and Fluoride, ppm	≤ 0.10
Suspended solids, ppm	≤ 0.050
Silica, ppm	≤ 0.10

a) Oxygen concentration in the makeup water to the RCS must not exceed 0.1 ppm when the reactor coolant temperature is greater than 180°F.

TABLE 9.2-11
THIS TABLE HAS BEEN DELETED.

TABLE 9.2-12
--DELETED--

TABLE 9.2-13
DECAY HEAT RATES[a]

TIME AFTER LOCA (SEC)	TOTAL HEAT RATE 10^6 BTU/HR
1×10^{-1}	1047.0
1×10^{0}	955.4
2×10^{0}	919.8
4×10^{0}	842.0
6×10^{0}	804.5
8×10^{0}	795.0
1×10^{1}	767.5
2×10^{1}	695.0
4×10^{1}	621.5
6×10^{1}	585.8
8×10^{1}	549.4
1×10^{2}	514.8
2×10^{2}	432.1
4×10^{2}	371.4
6×10^{2}	339.0
8×10^{2}	306.5
1×10^{3}	285.8
2×10^{3}	224.2
4×10^{3}	179.9
6×10^{3}	161.5
8×10^{3}	143.4
1×10^{4}	132.7
2×10^{4}	108.6
4×10^{4}	89.4
6×10^{4}	81.7
8×10^{4}	74.0
1×10^{5}	69.0
2×10^{5}	55.0
4×10^{5}	43.0
6×10^{5}	37.7
8×10^{5}	32.4
1×10^{6}	29.3
2×10^{6}	21.6
2.6×10^{6}	19.3

a) Total decay heat found by using the rated power level of 3458 Mwt

TABLE 9.2-14
THIS TABLE HAS BEEN DELETED.

9.3 PROCESS AUXILIARY

9.3.1 COMPRESSED AIR SYSTEMS

The Compressed Air System (CAS) consists of two separate systems, Instrument Air System (CI) and Service Air System (CA). The Instrument Air System is designed to provide a reliable supply of clean, dry, oil-free air of suitable quality and pressure for pneumatic instruments and controls and pneumatically operated valves for normal plant operation. The Service Air System is designed to provide the necessary compressed air for pneumatic tools and general plant usage. The CA, via a filtering skid, also provides suitable air to inflate the refueling lift gate seals.

9.3.1.1 Design Bases

The CAS serves no safety function because it is not required to achieve safe shutdown or to mitigate the consequences of a DBA. Therefore, the CAS is not a nuclear-safety-class-designed system.

However, the lines penetrating into containment are designed in accordance with NRC GDC 2, 4, 5, and 56 [1], [2], [3], [4], NRC Regulatory Guides 1.26 and 1.29 [5], [6], and applicable Branch Technical Positions [7], [8], as described in detail in Section 6.2.4, and ANSI N18.2 [9].

Where it is desired to maintain control of selected air-operated valves and dampers after safe shutdown, a safety related air accumulator and a double set of check valves are provided. For the purposes of design basis, functional, and testing requirements, this accumulator and its check valves are considered an extension of the component which it serves.

The air accumulators, the air piping (between the first check valve and the pneumatic control valve) and the associated valves are designated as ANS Safety Class 3 and designed in accordance with the requirements of ASME B&PV Code Section III, Class 3. In addition, they are designed to meet the requirements of seismic Category I in compliance with NRC Regulatory Guide 1.29. (See Tables 3.2-2 and 17A-1)

1. System Design

   The design of the CAS is based on an instrument air normal requirement per unit of 450 scfm (600 scfm with breathing air in service) with a transient maximum of 500 scfm (650 scfm with breathing air in service). The design of the CAS is also based on the service air maximum requirement of 1300 acfm for both units.

   The compressed air supplied has the following design bases:

   a. Instrument and service air regulates 80 to 120 psig nominal pressure

   b. Maximum instrument air dewpoint which meets the requirements of ANSI ISA S7.0.01 – 1996.

   c. Instrument air supplied to the system at the outlet of the dryer will be oil free and cleaned to remove particulate using one micron filters which are 98% or greater efficient.

d. Quantity of instrument air sufficient to provide for all pneumatic controls and valve and damper operators expected to operate simultaneously

e. Instrument air which is oil free of Grade “D” quality for use as breathing air

f. Quantity of service air sufficient for all equipment expected to operate simultaneously plus an allowance for the use of maintenance tools

g. Failure of the service air distribution system not to cause a loss of instrument air

h. Air systems piping is in accordance with ANSI B31.1, piping between Containment isolation valves is per ASME Section III, Code Class 2.

i. Quality of instrument air supplied to individual components will be consistent with manufacturer’s recommendations for air quality.

System operation is not required to initiate operation of engineered safeguards equipment because all air-operated valves and dampers are designed to fail-safe upon loss of control air. Air accumulators are provided for safety-related air-operated equipment where control is desired after safe shutdown.

System operation is not required to maintain the refueling gate seals (Figure 9.3-1) as described in section 9.1.4.2.3, item 16, because one bladder is always isolated from the instrument system (i.e., is passive) when the gate is closed.

2. Equipment Design

a. Instrument Air System

The equipment design parameters are given in Table 9.3-1. System components are located in the Electrical Building and Turbine Building. The main components have the following design characteristics:

1. Instrument Air Compressors

Six instrument air compressors are provided for both Units. Each compressor is a two stage, heavy duty, oil free, water cooled rotary air compressor. Each Lead train with it’s spare or back-up unit will supply dry, oil free air at 100 PSIG nominal pressure to meet the maximum instrument air requirement of each Unit. For both Units 1 and 2, the designation of which compressor is the lead and which is the backup is manually selectable.

Each instrument air compressor is provided with a dry filter-silencer in the intake line. The unit air compressors are cooled by the respective unit’s non-safety loop of the CCWS. The two common air compressors are cooled by the Turbine Plant Cooling Water (TPCW) system. Each of the Unit 2 and common air compressors have a dedicated trim cooler to provide supplemental cooling for the compressors. Non-safety chilled water is routed to the trim coolers to maintain CCW and TPCW cooling

temperatures within operating limits. The cooling water supply piping for the Unit 1 compressors in the Electrical Controls Building (ECB) is in series with the corresponding Unit 1 compressor in the TB allowing a single trim cooler to cool the water supply. The compressors located in the Turbine Building have an additional non-safety chilled water ventilation cooler that ducts cooled air into the compressor air inlet, ensuring intake parameters are within operational limits. The cooling water supply for the common compressor in the ECB is in parallel with the corresponding common compressor in the TB.

2. Instrument Air Aftercoolers

   The aftercoolers for the six rotary air compressors are an integral part of the compressor packages. Each aftercooler is provided with a moisture separator and a moisture trap for automatic removal of the separated water.

3. Instrument Air Receivers

   Six instrument air receivers are provided, one for each rotary air compressor.

4. Prefilters and Afterfilters

   Each air dryer package is provided with two prefilters and two afterfilters to remove moisture aerosol and solid particles that may be carried by the air.

5. Air Dryers

   Six air dryers of the heater type are provided, one for each rotary air compressor. The dryers are of the dual tower internally heated regenerative, desiccant type. The heating elements aid in desiccant regeneration. Each dryer has two towers charged with desiccant beads which adsorb a definite quantity of moisture vapor from the instrument air. During normal operation, one tower provides the drying or adsorption action while the other is being regenerated. Through a system of actuators, linkages, interconnecting piping, and valves, a continuous automatic alternating or cycling of the two towers is possible. Each air dryer is a completely packaged, self-contained unit, which is capable of drying the full instrument airflow requirements of one air compressor to -40°F dewpoint at a line pressure of 100 psig.

6. Piping

   Instrument air piping materials are as follows:

   (a) From the compressor area to the instrument air distribution header: stainless steel ASTM A312 Type 304 or 316

(b) From the distribution header up to the Containment penetration: copper tubing ASTM B88

(c) Between the Containment isolation valves: stainless steel SA312 Type TP304 or 316

(d) Piping inside the Containment and the instrument air yard piping: stainless steel ASTM A312 Type 304 or 316

(e) Tubing inside Containment, stainless steel ASTM A213, Type 316

b. Service Air System

The equipment design parameters are given in Table 9.3-2.

The common compressor is located west of the unit 1 Turbine Building. The main components have the following characteristics:

1. Service Air Compressor

One service air compressor is shared by both Units 1 and 2. It is a single stage, oil injected, air cooled rotary screw, 1300 ACFM air compressor. The compressor is sized to supply air at 100 psig normal pressure to meet the maximum service air requirements of both units.

2. Service Air Receivers

One service air receiver is provided for both units. The receiver is cylindrical and horizontal and is provided with an automatic moisture trap.

3. Piping

The service air piping material is carbon steel ASTM 106 Grade B. The piping code, when applicable, is ANSI B31.1, except for the Containment penetration which has an ANS Safety Class 2 classification and is designed to the requirements of the ASME B&PV Code, Section III, Code Class 2 as discussed in Section 6.2.4. Piping associated with accumulators (between the first check valve and the pneumatic control valve) is designated ANS Safety Class 3 and designed to meet ASME B&PV Code, Section III, Code Class 3.

9.3.1.2 System Description

The CAS is shown schematically on Figure 9.3-1 and Figure 9.3-2.

1. Instrument Air System

During normal system operation, instrument air to each unit is supplied by a train of equipment consisting of a lead instrument air compressor package, air receiver, and air dryer package. Also, a backup train automatically comes “on line” if the air header

pressure decreases. The lead trains for Unit 1 and 2 are manually selectable regarding which is the lead or backup train. Two complete common standby train packages are available (normally one common train is aligned to each unit) and automatically come “on line” if the unit compressors cannot maintain header pressure. The common compressors can be manually selected as the lead or backup compressor for either unit.

Before being delivered to the distribution network, the instrument air is filtered, dried, and filtered again. For this purpose, each air compressor is provided with a dedicated air dryer skid consisting of two 100% capacity prefilters, two 100% capacity air dryer towers and two 100% after filters.

The instrument air lines which penetrate the Containment are provided with two isolation valves. In addition, a flow-limiting orifice is provided for each penetration.

A pressure control valve and local supply station are provided for the refueling gates in the Fuel Building

Air-operated valves throughout the plant are arranged for safe failure in the absence of air.

Where it is desired to maintain control of selected air-operated valves and dampers after safe shutdown, an accumulator and a double set of check valves are provided.

The air piping between the first of the two check valves and the control valve is designed to seismic Category I requirements. The accumulators, in addition to meeting seismic Category I criteria, are designed to the requirements of the ASME B&PV Code, Section III, Code Class 3. The check valves are spring loaded and have resilient seats to ensure bubble-tight shutoff over the range of differential pressures to 120 psi.

Accumulators are provided for the auxiliary feedwater control valves in the individual feed lines and the motor-driven pumps recirculation valves which share accumulators with the flow control valves, the steam admission valves in lines supplying the auxiliary feedwater turbine, each Steam Generator PORV, and the CCW return pressure controller for the safety chillers. The accumulators at each auxiliary feedwater control valve are sized on the basis of allowing the operator remote manual control to isolate a faulted steam generator for a period of 30 minutes after loss of air. The basis for the recirculation valves is one valve cycle, then maintain the valve in the closed position for the remainder of the 30 minute period. See Section 10.4.9.2 for more details. The accumulators at the steam admission valves in lines supplying the auxiliary feedwater turbine are sized to have sufficient air capacity to drive the valve closed and maintain it closed for 7 hours post-accident plus an additional one-half hour for the AFWPT steam supply lines to then be locally-manually isolated by operator action, including any applicable air leakage criteria. The Steam Generator PORV accumulators are sized for Steam Generator Tube Rupture mitigation. They also provide motive power for hot standby and cooldown (See Section 5A) as long as the supply lasts followed by manual operator control to complete cooldown to hot shutdown condition. Each Steam Generator PORV accumulator can provide for 15 positionings over a four hour period after loss of instrument air. The CCW pressure control valves for the safety chillers have accumulators which are sized for 30 minutes of post accident operation.

Accumulators are provided for the control room air inlet control dampers. The dampers also have manual controls. This conservative sizing criteria, used to select tank volume for fabrication (to permit 10 operations of the dampers for a period of 30 days following the loss of the normal air supply system), ensures that the minimum system requirements are met. For the control room air inlet control dampers, the minimum system requirement is met if the air accumulator can hold the damper open on loss of instrument air until it is restored, the need for pressurization of the control room is past, or the operator can lock the damper open (not less than 2 hours after loss of instrument air). An evaluation of post-accident access to these dampers is provided in the Response to the NRC Action Plan Developed as a Result of the TMI-2 Accident, Section II.B.2.4.

2. Service Air System

During normal operation, service air to each unit is supplied by a service air compressor and a service air receiver. The service air is then delivered to the distribution system. The service air line which penetrates the Containment is provided with two isolation valves. In addition, a flow limiting orifice is provided for each penetration.

A pressure control valve and local supply station is provided for the refueling gate in each Containment Building.

9.3.1.3 Safety Evaluation

1. Instrument Air System

The Instrument Air System is not required for the safe shutdown of the plant, however, the Instrument Air System supplies air to the components important to safety. Therefore, all control and actuation devices associated with safeguards are electrically operated or, if pneumatic, have a fail-safe position. Certain valves are provided with local individual air accumulators. Accumulators are designed and sized to provide the required air quantity for a short period of time after loss of the main air system.

Although the Instrument Air System does not have a safeguard function, it is necessary for the operation of the plant, and the availability of air greatly facilitates the plant recovery following an emergency. For this reason, the unit compressors which are automatically tripped by a BOS will be automatically loaded after 90 seconds. An S signal will automatically trip the air compressors and dryers. The system is designed for high reliability, covering possible component failures with adequate backups and redundancies.

Generally, the accumulators are sized to ensure a minimum of one operating cycle of the valve (i.e., open - close - open) or to maintain the valve in the appropriate position until manual action can be provided (i.e., 30 min, but dependent on valve accessibility).

The valves and dampers which have accumulators to enable a safety function, are the auxiliary feedwater flow control valves and the motor-driven pumps recirculation valves which share accumulators with the flow control valves, the steam admission valves in lines supplying the auxiliary feedwater turbine, the Steam Generator PORV’s, and the Control Room air dampers. The valves and dampers fail in a safe position on loss of air.

2. Service Air System

The system is not nuclear-safety-related.

3. Failure Mode Analysis

The CAS is used to inflate the pneumatic seals on the refueling gates. The CAS and nitrogen bottles back-up are disconnected from at least one seal whenever a gate is in Seismic Category I service. Therefore, no failure or malfunction in the NNS CAS can affect the safety related functions of the refueling gates.

A failure mode and effects analysis for air-operated valves is given in Table 9.3-3. All valves within this table are safety-related in that they are components of safety-related systems and meet the applicable requirements of the ASME B&PV Code, Section III, which insures the pressure-retaining capability of the valve body. In addition, the valve actuator mechanisms are qualified for the service conditions (including seismic qualification for active valves) and are capable of controlled opening or closing against specified maximum pressure differentials and within specified time limits where required.

As indicated in Table 9.3-3 loss of instrument air does not jeopardize plant safety nor is operator action required to place the plant in a safe condition.

The results of the failure mode analysis demonstrate that a failure of the non-nuclear-safety-related CAS has no effect on plant safety because the valves fail to the specified positions by spring force alone. Valves the operator may wish to control beyond the specified failure position are equipped with air accumulators, as previously described, and listed in Table 9.3-3. The types of failure being analyzed are as follows:

a. Power Failures

The motor drivers for the six instrument air compressors are connected to six separate electrical buses. Four of the six air compressors are fed from class 1E buses to assure availability after an off-site electrical power failure. The remaining two machines (common spares) are fed from non-class 1E buses.

In the exceptional case where all electrical power is interrupted, the air compressors stop operating. Air-operated valves throughout the plant are arranged for safe failure in the absence of air. In this case, the valves are positioned to preserve the safety of plant and personnel.

b. Dryer Failures

The duplex dryer towers are arranged so that one regenerates while the other is in service. The two dryer towers interchange automatically on a time basis and an alarm at the main control room is provided to indicate a high differential pressure in any filter unit or failure in the microprocessing unit. In addition if a dryer for one of the lead rotary air compressors fails or malfunctions, the unit back-up train can be brought on line.

c. Filter Plug-Up

Normally, plant operators clean filters on a regular basis and switch them into service before plugging can occur. In the exceptional case that a filter does plug, a differential pressure alarm warns the operator, who manually shifts the airstream to the second filter.

d. Air Header Ruptures

A separate header is provided for each CAS and rupture of one header leaves the other headers unaffected. As with a power failure discussed in Subsection 9.3.1.3, Item 3.a, if a rupture of the instrument air header occurs, the plant can be shut down in a safe manner. Air headers are routed to prevent affecting safety-related equipment in the event of a rupture.

e. Equipment Sharing

Both units share a service air compressor. Only the common spare instrument air compressors can be aligned to either unit. Failure of the common service air compressor will not impact a safe shutdown or mitigate the consequences of a DBA.

9.3.1.4 Tests and Inspections

The systems are inspected and cleaned prior to service. Instruments are calibrated during testing, and automatic controls are tested for actuation at the proper set points. Alarm functions are checked for operability and limits during plant operational testing. The system is operated and initially tested with regard to flow paths, flow capacity, and mechanical operability.

9.3.1.5 Instrumentation Requirements

The instrument air compressors are cycled to maintain the set desired pressures. Both Unit 1 and 2 lead compressors are cycled on/off based on a discharge air pressure switch integral to the packages. The backup air compressors are cycled on/off based on an abnormally low pressure in either of their designated air receivers. It is anticipated that during normal operation the lead air compressor will run continuously with the backups cycling on/off as needed. The pressure at the air header is indicated locally and in the Control Room.

9.3.2 PROCESS SAMPLING SYSTEM

9.3.2.1 Design Bases

A separate sampling system is furnished for each unit. Each system is designed to provide liquid and gas samples at controlled temperatures and pressures for laboratory analysis. The samples are drawn from the points designated in Table 9.3-4. To ensure representative samples, samples are taken from turbulent flow zones, i.e., downstream of pump discharges or in tank recirculation lines. Pumped recirculation of tank contents and sample withdrawal from recirculation lines avoids sampling from low points or potential sediment traps. These samples are routed to a central location where the components of the Process Sampling System are located. The fluid properties at the designated sample points are critical to proper functioning of the plant and require frequent testing, as detailed in Technical Specifications.

Local sampling is performed where batch processing is required, i.e., Liquid Waste Processing System (LWPS), Gaseous Waste Processing System (GWPS), Boron Recycle System (BRS), and for plant effluent streams. In addition, local samples can be taken from the boric acid tanks, boric acid batching tank, Refueling Water Storage Tank, and the chemical additive tank. Refer to applicable Flow Diagrams. In general, the local sampling points require less frequent testing and are therefore not routed to the Process Sampling System. Local and effluent sampling is discussed in Section 11.5.4.

The Process Sampling System (PSS) may be used to obtain samples post-accident. During a loss-of-coolant accident (LOCA), PSS sample lines are isolated on both sides of the Containment boundary. See Section 6.2.4 and II.B.3 for PSS sample line isolation details.

9.3.2.2 System Description

9.3.2.2.1 System Operation

Samples from the points designated in Table 9.3-4 are conveyed to the sampling room in stainless steel tubing. The line sizes are determined to meet analyzer and grab sample flow requirements. The distance between the sample point and Process Sampling System has been kept to a minimum to reduce possible plateout and precipitation.

Samples enter the sample conditioning panel which houses the sample coolers, pressure-reducing and pressure-regulating valves, and local pressure and temperature indicators. The requirements for conditioning of a sample are based on the design temperature and pressure at the point of sampling. After conditioning, sample temperature and pressure are less than 120°F and 5 to 80 psig max., respectively.

All the sample lines are provided with grab sample valves or sample vessels with quick disconnect fittings, as shown on the Primary Sampling System piping and instrumentation diagram (P&ID), Flow Diagram M1(2)-0228.

Automatic online analyzers for cation conductivity, high and low pH specific conductivity, and sodium ion concentration are provided for the steam generator blowdown samples. An inline radiation monitor is provided for the steam generator blowdown samples to indicate steam generator tube leakage.

The samples are segregated so that primary and secondary fluids are not mixed. The routing of each sample line is shown on the Primary Sampling System P&ID, Flow Diagram M1(2)-0228.

Automatic online analyzer provided for the CVCS Letdown Upstream of Purification Demineralizer sample to measure RCS dissolved hydrogen and oxygen.

9.3.2.2.2 Components Description

1. Samples Coolers

   The samples coolers are of tube-in-shell design with pressure relief valves on the shells. The samples flow through the tube side and component cooling water flows through the shells. The sample cooler tubes are of stainless steel construction and the shells are carbon steel.

2. Pressure-Reducing and Pressure-Regulating Equipment

The pressure-reducing and pressure-regulating equipment consists of a variable capillary device or a velocity-controlled pressure-reducing valve in series with a pressure-regulating valve to maintain sample stream pressure at 5 to 80 psig max. downstream of the regulating valve. Relief valves are provided on each sample line as an additional safety feature. The relief valves discharge to the waste holdup tank or to an appropriate drain system as shown on Flow Diagram M1(2)-0228.

3. Sample Vessels

Sample vessels or bombs are used for all gas sampling and the reactor coolant, pressurizer, and RHR System liquid samples. The vessels are of stainless steel construction designed to withstand the reactor coolant design pressure and temperature.

4. Sample Sink

Each sample sink is located in a hooded enclosure equipped with an exhaust ventilator ducted to the Containment vent header. The sinks and enclosures are of stainless steel construction and have raised edges to minimize liquid spillage.

9.3.2.2.3 Manual Sampling Techniques

1. Liquid Sampling

Liquid sampling is accomplished by purging the sampling lines. This allows the sample to flow for a sufficient time to ensure that representative samples are obtained.

2. Sample Vessel

Sample vessels, used to collect gaseous samples and certain liquid samples, are used by first adequately purging the sample vessels and then closing inlet and outlet vessel valves. The sample vessel is then removed from its connection lines after the sample flow has been stopped.

3. Sample Line Purging

Sample line purging is a function of the sample line diameter and distance to the point of sampling. Sample line purge flow rates are recommended not to be less than those values as identified in Table 9.3-4.

9.3.2.2.4 Codes and Standards

The equipment in the sampling is designed, furnished, and installed to meet applicable codes and standards. NRC Regulatory Guides 1.21, 1.26, and 1.29 and ANSI N13.1-1969 were applied to the design of the Process Sampling System as follows:

1. NRC Regulatory Guide 1.21

   Monitoring of effluents is not a function of the Process Sampling System. Effluent sampling is discussed in Section 11.5.

2. NRC Regulatory Guide 1.26

   All valves, piping, and tubing originating within the Containment are designed to meet the requirements of ASME B&PV Code, Section III, Class 2. All valves and tubing downstream of the outside Containment isolation are non-nuclear-safety class.

3. NRC Regulatory Guide 1.29

   All valves, piping and tubing located inside the Containment and outside the Containment up to the second isolation valve (the safety related portion of the system) are designed to the requirements of the seismic Category I. All sample lines (including valves) downstream of the second isolation valve (outside the containment) leading to the sampling panels (the non-safety related portion of the system) are designated as Class 5.

   A Class 1E panel is provided to operate the sample line Containment isolation valves from the hallway outside the sample room.

4. ANSI N13.1

   Monitoring of airborne radioactive materials inside the plant and in effluents is not a function of the Process Sampling System. Section 12.3.4 discusses this subject.

9.3.2.3 Safety Evaluation

A separate sampling system is furnished for each unit so that there is no sharing of equipment between units.

The operation of the sampling system is not necessary for safe plant shutdown after an accident. Therefore, in the event of an accident, all sample lines which pass through the Containment are automatically isolated by fail-closed air-operated valves on either side of the Containment. The Containment Isolation System is described in Section 6.2.4.

Similarly, no alternate paths to the Process Sampling Systems during accident conditions are required because the system cannot cause loss of function of active components whose operation is necessary for safe shutdown of the plant.

The routing of high-temperature and high-pressure sample lines outside the Containment is not considered hazardous because of their limited flow, intermittent use, and nonessential nature. Also, the location of sample temperature-reduction and pressure-reduction equipment outside the Containment is considered beneficial from a components maintenance standpoint. Passive flow restrictions are not provided because of intermittent use and the small size (3/8 in.) of the sample lines. The use of hot leg sample line taps with a 0.234" bore provides flow restriction to limit reactor coolant loss from a rupture of a sample line in compliance with the intent of Standard Review Plan Section 9.3.2, Acceptance Criterion 2.g. For further clarification, see FSAR Section 5.4.3.2 and Figure 5.1-1.

The reactor coolant loop samples are passed through 3/8 inch tubing; configured to act as delay coils located inside the Containment to permit the decay of short-lived isotopes. Samples lines are shielded, as required, to protect plant personnel from radiation in the course of their normal duties. Radioactivity releases resulting from break of a reactor coolant sample line are discussed in Section 15.6.2.

9.3.2.4 Testing and Inspection

Each component is inspected and cleaned prior to installation. Instruments and analyzers are calibrated during testing. Automatic controls are tested for actuation at the proper set point. The system is operated and tested initially with regard to flow paths, flow capacity, thermal capacity, and mechanical operability.

9.3.2.5 Instrumentation Application

Local temperature and pressure indicators are provided on all high-temperature and high-pressure sample lines. These local instruments enable the operator to determine if the sample coolers and pressure-regulating equipment are functioning properly. Local flow indicators are used to manually adjust the flow of samples to analyzers and to the closed sample vessels as shown on the Flow Diagram M1(2)-0228.

Radioactivity in the steam generator blowdown fluid is measured by a single inline radiation monitor which sees a combined sample from the four steam generators. If the monitor alarms, the blowdown and blowdown sample outboard Containment isolation valves and inboard process isolation valves close. These valves are reopened from the Control Room and the samples can pass through the monitor one at a time by manually operating local valves to ascertain which steam generator is leaking.

Cation conductivity, pH specific conductivity, and sodium ion concentration are continuously monitored for each steam generator. These parameters guide the selection of chemical quantities to be fed to the condensate and the regulation of steam generator blowdown flow rate. (See Sections 10.3.5 and 10.4.8, respectively.) By using the monitors in conjunction with proper chemical treatment and sufficient blowdown flow, the steam generator water chemistry can be kept within prescribed limits, as detailed in Section 10.3.5. High conductivity, high pH and high sodium levels are alarmed on a local annunciator, which has a common alarm relay to the Control Room.

Specific conductivity is used to monitor the performance of the blowdown cleanup demineralizers.

An electrical panel for operating the Containment isolation valves on the sample lines is provided in the hallway outside the sample room. By first aligning valves on the sample conditioning panel and the sample hood, the operator can then open the appropriate Containment isolation valves to draw a sample. Isolation valves are of the fail-closed type and close automatically on Containment isolation signal.

9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

9.3.3.1 Design Bases

Liquid wastes, valve and pump leakoffs, tank overflows, and tank drains are collected by the equipment and floor drainage system. Drains are separated according to their activity and quality as follows [10]:

1. Aerated tritiated drains of reactor coolant quality are collected in a waste holdup tank for treatment, recycling or disposal (Drain channel A subsystem).

2. Tritiated and nontritiated drains from the Containment sumps, Auxiliary Building sumps, Auxiliary Building and Safeguards Building floor drains, and laboratory drains, as outlined in Section 11.2, are collected in the floor drain tank for treatment, recycle, or disposal (Drain channel B subsystem).

3. Laundry and hot shower wastes and decontamination wastes are collected in the laundry and hot shower tank for treatment, recycle, or disposal (Drain channel C subsystem).

4. Tritiated aerated radioactive leakage outside the Containment from valve and pump seal leakoffs, recycle holdup tank overflows, and other equipment is routed through the equipment drain header to the waste holdup tank for treatment, recycle, or disposal.

5. Tanks containing potentially radioactive fluids have high-level alarms or level indicators as shown in Table 9.3-5. Therefore, the operator is alerted prior to an overflow condition. Any overflows are collected as shown on Table 9.3-5.

6. Provisions are made for routing relief valve discharges from potentially radioactive fluid tanks to the Liquid Waste Processing System. These discharges are collected as shown on Table 9.3-5.

   Monitoring and sampling of potentially radioactive fluids is described in Section 11.5.

   The drainage system used to carry clean water is completely separate from the drainage system used to carry potentially radioactively contaminated water to prevent the accidental release of radioactive material to the environment. Equipment drains are also provided in some areas in addition to floor drains.

   Turbine Building floor drains, which normally handle clean water but can potentially carry contaminated water caused by equipment failures or accident conditions, are provided with radiation monitors which detect radioactive contamination before discharging the drains to the environment.

   Sumps, tanks, and pumps are located at lower levels to facilitate transfer to the Waste Processing System (WPS). For radiological considerations of the WPS under normal operating conditions and postulated spills and accidents, see Chapters 11, 12, and 15.

   The essential portions of this system are those drains serving areas with safety-related equipment. The floor drainage system serving areas with equipment required for the safe shutdown of the plant is designed to handle postulated equipment gross leakages.

The floor drainage system serving areas with safety-related equipment and components is segregated into independent systems where a postulated flooding of one area does not impair the safety of the redundant equipment.

9.3.3.2 System Description

Segregated drain headers are provided for each level of the Auxiliary Building and Containment, as described in Section 11.2.

Collection and transfer of liquids from the lower levels of buildings are facilitated by sump pumps, sumps, and drain collection tanks. Spare, full-capacity sump pumps are provided at each transfer point in the drainage system.

Backwater valves are provided, as necessary, to prevent water from backing up through a drain line and possibly impairing the function of safety-related equipment. Ball Float devices, which allow water to drain from an area but prevent steam propagation or unfiltered air inleakage to other interconnecting areas, are provided as necessary for the protection of essential systems and components due to postulated piping failures per FSAR Section 3.6B and for control room habitability during a DBA. For the equipment and floor drain system flow diagrams, see Figures 9.3-5, 9.3-6, 9.3-7, 9.3-8, and 9.3-9.

9.3.3.2.1 Containment Building Floor Drains

Leakage from all Containment Building floors is directed to one of the Containment sumps. The reactor cavity is also provided with a sump at the lowest point in the Containment Building. The sump pumps from all three sumps are typically aligned with its respective floor drain tank. These sumps can be alternately aligned to the waste holdup tank.

An increase in the leakage rate inside the Containment can be detected by the following:

1. An increase in airborne contamination

2. An increase in condensate from the Containment cooling units

3. An increase in humidity in the Containment air

4. An increase in the frequency and/or run time of sump pumps in containment.

For more information about Containment leakage detection within the Containment, see Section 5.2.5.

The Containment Building drains do not function during an accident and are isolated by two fail-closed, air-operated Containment isolation valves, one inside and one outside the Containment. The Containment Building drain piping is non-safety class piping except for the Containment penetration piping, and both Containment isolation valves which are ANSI N18.2 Safety Class 2 [5], [9].

9.3.3.2.2 Safeguards Building Floor Drains

Engineered safeguard pumps are located on the lower elevation of the Safeguards Building. The pumps are divided into two redundant trains with each located in an area segregated from its backup train by a watertight wall, preventing a total flooding of one area from interfering with the operation of the pumps located in the other area.

Each area at or below 785 ft 6 in. is provided with a sump, independent from the sump serving the other area. These areas have floor drain headers discharging into the closed Safeguards Building sumps. Drains from areas above 785 ft 6 in. are routed directly to the floor drain tank or to one of the sumps. The Diesel Generator Room sump pumps are normally aligned for discharge into the Wastewater Management System, for treatment prior to discharge.

CCW Tank Room Sump No. 3, which collects drainage from the component cooling water drain tank, Containment spray and RHR pumps, is aligned to discharge into the Wastewater Management System. CCW Tank Room Sump No. 3 is only fed by drains which collect fluid from normally non-radioactive sources.

Safeguards Building floor drain sumps 1 and 2 (for each unit) are normally aligned for discharge into the floor drain tank and have an alternate line to the waste holdup tank. The floor drains from the floor drain tank (no. 1 for Unit 1 and no. 2 for Unit 2) and floor drain tank room utilize a locked closed valve to prevent recirculation of the tank contents back into Sump No. 1 in the event of failure of the tank or piping within the tank cubicle.

Piping and pumps required to prevent flooding of safety-related equipment are ANSI N18.2 Safety Class 3 [5], [9]. All other equipment is non-nuclear-safety class.

9.3.3.2.3 Auxiliary Building Floor Drains (ABFD)

Leakage from the Auxiliary Building is collected in various sumps or to gravity feed drains to the floor drain tanks. Drainage from the Battery Room, Secondary Sampling Room, Chemical Storage Room, Air Compressor Room, Mechanical Equipment Room, and Cable Room is routed to floor drain Sump 11. The ABFD Sump 11 is normally aligned to discharge to the Wastewater Management System. ABFD Sumps 3 and 10 receive drainage from the component cooling water heat exchangers and safety related chillers, respectively. ABFD Sump 10 is normally aligned to discharge to the Unit 1 Component Cooling Water Drain Tank, with an alternate connection to the Unit 2 Component Cooling Water Drain Tank. ABFD Sump 3 pump is normally aligned to discharge directly to the LVW System. ABFD Sumps 4, 5, and 9 receive drainage from the southern half of the Auxiliary Building below elevation 792 ft 0 in. ABFD Sumps 4, 5, and 9 are aligned to discharge to floor drain tank No. 1. The northern half of the Auxiliary Building is drained into various sumps. ABFD Sumps 6 and 8 take drainage from the waste evaporator condensate tank and recycle evaporator feed pump areas, respectively. In addition, ABFD Sump 8 takes drains from a portion of Unit 2 turbine building elevation 810 ft. 6 in. ABFD Sumps 6 & 8 are aligned to discharge into the waste holdup tanks. ABFD Sumps 1, 2, 7, and 12 collect the remainder of the drainage from the northern half of the Auxiliary Building. ABFD Sumps 1, 2, 7, & 12 are typically aligned to discharge to floor drain tank No. 1. ABFD Sump 2 has an alternate connection to Floor Drain Tank #3.

Discharges from ABFD Sumps 1, 7, and 12 have an alternate connection to floor drain tank 2.

Boron recycle tanks and waste holdup tanks are located within watertight compartments to prevent leakage into other areas if a tank break occurs. Floor drains from these areas have a locked-closed, manually operated gate valve located outside each compartment.

9.3.3.2.4 Turbine Building Floor Drains

In the Turbine Building, the bulk of the drainage is collected in the two Turbine Building sumps. The Turbine Building sumps are normally aligned to eventually discharge to the Low Volume Waste (LVW) Pond of the Wastewater Management System (WMS), for conventional treatment prior to discharge. However, in the event of radioactive contamination in the sump discharge, in excess of limits established in the Radiological Effluent Controls Program, release to the LVW Pond will be automatically diverted to the Co-Current Waste (COW) Treatment System, for conventional and radiological treatment. A leak of radioactive material into Turbine Building sump water is detected and alarmed in the Control Room. The radiation level of Turbine Building sump water can be monitored from the Control Room. For information on radioactive leakage, see Section 11.2.

Other means of drainage in the Turbine Building are as follows:

1. Drains in the hot and cold labs drain by gravity to the floor drain tank for Unit 1. The sinks in the chemistry instrument lab drain by gravity to the floor drain tank for Unit 1.

2. Drains in the personnel decontamination and chemical instrument lab drain by gravity to the laundry and hot shower drain tank.

3. Leakoffs for various secondary components drain to the atmospheric drain tank. The tank is drained to the main condenser spray headers or main condenser gravity drains. Overflow from the tank is drained to Turbine Building sump No. 2.

   All drainage piping and equipment in the Turbine Building are non-nuclear-safety-related and are not required to function after an accident or for safe reactor shutdown.

9.3.3.2.5 Fuel Building Floor Drains

All drainage from the Fuel Building is directed into one of two sumps. Most drainage from the Fuel Building is directed to Fuel Building sump No. 1, which is located at elevation 800 ft 2 in. Fuel Building Sumps 1 and 2 are aligned to discharge to floor drain tank No. 1, which is located in the Safeguards Building at elevation 773 ft 0 in.

9.3.3.3 Safety Evaluation

Each compartment that houses safety-related components is designed to accommodate a design basis flood resulting from a 50 gpm leak for 30 minutes in any one compartment. The highest resulting flood level is 28 inches above the floor for the centrifugal charging pump rooms. All components important to safety are installed above the level resulting from the postulated flood. Thus, the maximum flood level will not affect operation of the safety-related electrical equipment in the room.

Also, since none of the backwater valves are seismically qualified, it is assumed that more than one valve will fail as a result of a safe shutdown earthquake. If communication between rooms

via the floor drain system is also assumed, the postulated flood (backflow) will be distributed over a wider area (multiple compartments) with a resultant flood level lower than if the flood level was confined to one compartment only and, as a result, be in acceptable levels in individual compartments. Therefore, the safety level of the plant will not be affected.

Although some backwater valves are credited to operate for the mitigation of postulated piping failures per FSAR Section 3.6B, a seismic event is not assumed in the analysis of these piping failures.

Compartments containing engineered safeguards pumps, recycle holdup tanks, and waste holdup tanks are designed to seismic Category I requirements [6].

Except for specific LOCA conditions, all portions of the Equipment and Floor Drainage System in the Safeguards Building that are at 773' and are safety related are operative at all times. Continued operation of the Containment portion of the system during a LOCA is not required and is prevented by automatic closure of the Containment drain piping isolation valves.

Main drain headers are at least four in. in diameter. Header size is sufficient to minimize total flooding of the piping and to ensure that the drain lines are not pressurized and ruptured [7], [8].

Floor drains and sump pumps in the lowest elevation of Safeguards Building are designed to accommodate a leakage of 50 gpm without flooding of adjacent areas.

Each sump is equipped with two pumps, either of which can handle the design leakage rate. This leakage is postulated to occur as a result of a gross flange gasket failure or a severely damaged pump seal.

The compartments housing engineered safeguards pumps In the Safeguards Building are designed to contain a leakage rate of 50 gpm for 30 min without causing flooding of adjacent areas, assuming all floor drains from this area are clogged. This design protects safety-related equipment outside the pump compartment from flooding due to leakage.

The Auxiliary Building floor drainage system is designed to accommodate normal expected leakage without localized flooding.

If an accident occurs, i.e., a rupture of a tank, a crack in large piping, and so forth, total flooding of a floor is prevented by the following:

1. Operator action to isolate the affected system

2. An arrangement of floor drains ensuring that an increased number of floor drains are used as the flooding expands

The combination of these two actions ensures that deep flooding of a building floor does not occur.

If a leak becomes large enough or is undetected for such a time that a floor is flooded, the water will begin to escape to the Auxiliary Building lowest elevation through stairways and so forth. However, with the exception of the component cooling water heat exchangers and the emergency chillers, no safety-related equipment is located on this floor.

A partial outside flooding of these heat exchangers does not impair their operability or integrity. The emergency chillers are located In a separate compartment under the Control Room, where they are not subjected to leakage from equipment at higher elevations.

The boron recycle and waste holdup tanks are located in separate watertight compartments. Each compartment is drained to the Auxiliary Building sumps, with locked-closed valves in each line outside the compartments. As a result of the administrative control placed on the valve, the possibility of leaving the valve inadvertently open is not considered credible.

Overflow or drainage of a tank is detected by the tank high-low level instrumentation. Except as indicated on Table 9.3-5, all potentially radioactive overflows are routed to receiving tanks within the liquid waste disposal system, thereby ensuring that sufficient holdup and monitoring capabilities are available prior to the ultimate disposal to the environment or recycling, as applicable [10].

9.3.3.4 Testing and Inspection Requirements

Leaktightness, flow capacity, and flow paths are tested prior to the initial operation of the system. Pumps and level controls are adjusted for maintenance of proper sump levels.

9.3.3.5 Instrumentation Requirements

Each safeguards area sump collecting the floor drainage from one Containment spray pump compartment, one safety injection pump compartment, one RHR pump compartment, and so forth has high-1, high-2, and high-high (high-1 < high-2 < high-high) level alarms in the Control Room to indicate that water is entering the sump. Upon receiving the high-1 signal, one sump pump begins to operate. If the level continues to increase to the high-2 level, the second pump starts. The operational status and running time of the sump pumps are indicated on the local panel. A high-high level alarm alerts the operator if the sump level continues to increase with both sump pumps in operation. Based on this information and the sump pumps design flow, the operator can approximate the leakage inflow to each sump.

The Auxiliary Building sump and the diesel generator floor drain sump have a high-water alarm, which alerts the Control Room operator. The operational status and running time of these pumps are indicated on the local panel.

The Containment sump pump discharge lines have a flow totalizer to indicate total flow as part of the overall Containment leakage control.

All sump pumps operate automatically; i.e., the start of the pumps is controlled by the sump levels.

As shown In Table 9.3-5, storage tanks which contain potentially radioactive fluids are provided with a high-water level alarm and level indication.

Refer to Section 9.2.3 for a discussion of the RMWST, Section 9.2.6 for the Condensate Storage Tank, and Section 11.2 and Table 11.2-1 for the liquid waste processing tanks.

9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM (INCLUDING BORON RECYCLE SYSTEM)

9.3.4.1 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS), shown in Figures 9.3-10 and 9.3-11 is designed to provide the following services to the Reactor Coolant System (RCS):

1. Maintenance of programmed water level in the pressurizer, i.e., maintain required water inventory in the RCS.

2. Maintenance of seal-water injection flow to the reactor coolant pumps.

3. Control of reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration and makeup.

4. Emergency core cooling (part of the system is shared with the Emergency Core Cooling System).

5. Provide means for filling, draining and pressure testing of the RCS.

9.3.4.1.1 Design Bases

Quantitative design bases are given in Table 9.3-6 with qualitative descriptions given below.

9.3.4.1.1.1 Reactivity Control

The CVCS regulates the concentration of chemical neutron absorber (boron) in the reactor coolant to control reactivity changes resulting from the change in reactor coolant temperature between cold shutdown and hot full-power operation, burnup of fuel and burnable poisons, buildup of fission products in the fuel, and xenon transients.

Reactor Makeup Control

1. The CVCS is capable of borating the RCS through either one of two flow paths and from either one of two boric acid sources.

2. The amount of boric acid stored in the CVCS always exceeds that amount required to borate the RCS to cold shutdown concentration assuming that the control assembly with the highest reactivity worth is stuck in its fully withdrawn position. This amount of boric acid also exceeds the amount required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

Boron Thermal Regeneration

The CVCS was designed to control the changes in reactor coolant boron concentration to compensate for the xenon transients during load follow operations without adding makeup for either boration or dilution. To accomplish this, the boron thermal regeneration process was designed to allow load follow operations as required by the design load cycle. Load follow operation is currently not utilized at Comanche Peak.

Because load follow operation is currently not utilized, there is no need for chilled water to be provided to the Letdown Chiller Heat Exchanger. Therefore, the BTRS chiller and associated components in the chilled water loop are normally removed from service and drained so as not to provide chilled water to the shell side of the Letdown Chiller Heat Exchanger. Also because load follow operation is currently not utilized, the heating source for the letdown reheat heat exchanger, CVCS water flow through the tube side of the heat exchanger, is normally isolated and the tube side of the heat exchanger is drained. There is no near-term plan to utilize these components for load follow operation. If the BTRS chiller and associated components are ever returned to service, flow to and from the thermal regeneration demineralizers and piping vibration will be verified prior to the return to service.

The BTRS is currently used to reduce reactor coolant boron concentration at the end of the core cycle. Additionally, the BTRS demineralizers may be loaded with various types of resin in order to provide supplemental or enhanced cleanup of CVCS letdown flow.

When the BTRS is placed in service, BTRS water taken from the RCS may flow through the tube side of the Letdown Chiller Heat Exchanger, as well as through the shell side of the Letdown Reheat Heat Exchanger. There is the possibility for leakage of water containing radioactivity from the tube side into the normally drained shell side of the Letdown Chiller Heat Exchanger, or from the shell side into the normally drained tube side of the Letdown Reheat Heat Exchanger.

Leakage into the shell side of the Letdown Chiller Heat Exchanger passes through to the Chiller Surge Tank. The high level alarm of the Chiller Surge Tank provides in-leakage detection when the BTRS is in service. Leakage water entering the tube side of the Letdown Reheat Heat Exchanger passes through open drain valves into floor drains which direct the water to Floor Drain Tanks 1 and 2, both of which have high level alarms. Tank overflow protection is further addressed in Table 9.3-5, item 9 of Section 11.2.2.4.2, and Section 11.2.2.5.2.

9.3.4.1.1.2 Regulation of Reactor Coolant Inventory

The CVCS maintains the coolant inventory in the RCS within the allowable pressurizer level range for all normal modes of operation including startup from cold shutdown, full power operation and plant cooldown. This system also has sufficient makeup capacity to maintain the minimum required inventory in the event of minor RCS leaks (see the Technical Specifications for a discussion of maximum allowable RCS leakage).

9.3.4.1.1.3 Reactor Coolant Purification

The CVCS is capable of removing fission and activation products, in ionic form or as particulates, from the reactor coolant in order to provide access to those process lines carrying reactor coolant during operation and to reduce activity releases due to leaks.

9.3.4.1.1.4 Chemical Additions for Corrosion Control

The CVCS provides a means for adding chemicals to the RCS which control the pH of the coolant during initial startup and subsequent operation, scavenge oxygen from the coolant during startup, and counteract the production of oxygen in the reactor coolant due to radiolysis of water in the core region.

The CVCS is capable of maintaining the oxygen content and pH of the reactor coolant within limits specified in Table 5.2-5.

9.3.4.1.1.5 Seal Water Injection

The CVCS is able to continuously supply filtered water to each reactor coolant pump seal, as required by the reactor coolant pump design.

9.3.4.1.1.6 Hydrostatic Testing of the Reactor Coolant System

The CVCS is capable of supplying water at the maximum test pressure specified to verify the integrity of the RCS. The hydrostatic test is performed prior to initial operation and as part of the periodic RCS inspection program.

9.3.4.1.1.7 Emergency Core Cooling

The centrifugal charging pumps in the CVCS also serve as the high-head safety injection pumps in the Emergency Core Cooling System. Other than the centrifugal charging pumps and associated piping and valves, the CVCS is not required to function during a loss of coolant accident (LOCA). During a LOCA, the CVCS is isolated except for the centrifugal charging pumps and the piping in the safety injection path.

9.3.4.1.2 System Description

The CVCS is shown in Figure 9.3-10 (piping and instrumentation diagram) with system design parameters listed in Table 9.3-6. The codes and standards to which the individual components of the CVCS are designed are listed in Section 3.2. The CVCS consists of several subsystems: the Charging, Letdown and Seal Water System; the Reactor Coolant Purification and Chemistry Control System; the Reactor Makeup Control System; and the Boron Thermal Regeneration System.

9.3.4.1.2.1 Charging, Letdown and Seal Water System

The charging and letdown functions of the CVCS are employed to maintain a programmed water level in the RCS pressurizer, thus maintaining proper reactor coolant inventory during all phases of plant operation. This is achieved by means of continuous feed and bleed process during which the feed rate is automatically controlled based on pressurizer water level. The bleed rate can be chosen to suit various plant operational requirements by selecting the proper combination of letdown orifices in the letdown flow path.

Reactor coolant is discharged to the CVCS from a reactor coolant loop cold leg; it then flows through the shell side of the regenerative heat exchanger where its temperature is reduced by heat transfer to the charging flow passing through the tubes. The coolant then experiences a large pressure reduction as it passes through the letdown orifice(s) and flows through the tube side of the letdown heat exchanger where its temperature is further reduced. Downstream of the letdown heat exchanger a second pressure reduction occurs. This second pressure reduction is performed by the low pressure letdown valve, the function of which is to maintain upstream pressure thus preventing flashing downstream of the letdown orifices.

The coolant then flows through one of the mixed bed demineralizers. The flow may then pass through the cation bed demineralizer which is used intermittently when additional purification of the reactor coolant is required. Two mixed bed demineralizers may be used in parallel during shutdown for maximum purification.

Alternatively, the flow may be directed such that the mixed bed demineralizers are bypassed. Flow could then be directed to the cation bed demineralizer and/or to the BTRS demineralizer(s) for purification process.

From a point upstream of the Boron Thermal Regeneration System or from a point upstream of the reactor coolant filters, a small sample flow may be diverted from the letdown stream to the Boron Concentration Measurement System. The Boron Concentration Measurement System is abandoned in place.

During reactor coolant boration and dilution operations, especially during load follow, the letdown flow leaving the demineralizers may be directed to the Boron Thermal Regeneration System. The coolant then flows through the reactor coolant filter and into the volume control tank through a spray nozzle in the top of the tank. Hydrogen (from the Gaseous Waste Processing System) is continuously supplied to the volume control tank where it mixes with fission gasses which are stripped from the reactor coolant into the tank gas space. The contaminated hydrogen is vented back to the Gaseous Waste Processing System. The partial pressure of hydrogen in the volume control tank determines the concentration of hydrogen dissolved in the reactor coolant for control of oxygen produced by radiolysis of water in the core.

Three pumps (one positive displacement pump and two centrifugal charging pumps) are provided to take suction from the volume control tank and return the cooled, purified reactor coolant to the RCS. Normal charging flow is handled by one of the three charging pumps. This charging flow splits into two paths. The bulk of the charging flow is pumped back to the RCS through the tube side of the regenerative heat exchanger. The letdown flow in the shell side of the regenerative heat exchanger raises the charging flow to a temperature approaching the reactor coolant temperature. The flow is then injected into a cold leg of the RCS. Two charging paths are provided from a point downstream of the regenerative heat exchanger. A flow path is also provided from the regenerative heat exchanger outlet to the pressurizer spray line. An air operated valve in the spray line is employed to provide auxiliary spray to the vapor space of the pressurizer during plant cooldown. This provides a means of cooling the pressurizer near the end of plant cooldown, when the reactor coolant pumps, which normally provide the driving head for the pressurizer spray, are not operating.

A portion of the charging flow is directed to the reactor coolant pumps (nominally 8 gpm per pump) through a seal water injection filter. It is directed down to a point between the pump shaft bearing and the thermal barrier cooling coil. Here the flow splits and a portion (nominally 5 gpm per pump) enters the RCS through the labyrinth seals and thermal barrier. The remainder of the flow is directed up the pump shaft, cooling the lower bearing, and to the number 1 seal leakoff. The number 1 seal leakoff flow discharges to a common manifold, exits from the Containment, and then passes through the seal water return filter and the seal water heat exchanger to the suction side of the charging pumps, or by alternate path to the volume control tank. A very small portion of the seal flow leaks through to the number 2 seal. A number 3 seal provides a final barrier to leakage of reactor coolant to the Containment atmosphere. The number 2 leakoff flow is discharged to the reactor coolant drain tank in the Liquid Waste Processing System.

This seal injection flow is assured because there are 3 charging pumps, which individually are capable of providing the normal charging line flow plus the nominal seal injection flow. A number 3 seal provides a barrier to leakage of reactor coolant to the Containment Atmosphere. The number 3 seal injection is supply by Reactor Makeup Water to a stand pipe. Part of the water flows downward, joins with the leakoff from the No. 2 seal and exits through the No. 2 seal leakoff. The remaining water exits through the No. 3 seal leakoff which is discharged to the Containment Sump.

In the unlikely event of a loss of seal injection flow to the RCPs, the Component Cooling Water System (CCWS) continues to provide flow to the thermal barrier heat exchangers. Under these conditions, the thermal barrier heat exchangers, functioning in its backup capacity, cools the reactor coolant before it enters the pump radial bearing and the shaft seal area. The loss of seal injection flow may result in an increase in the number 1 seal leak rate, a temperature increase in the pump bearing area, and a temperature increase in the seal area. If seal injection flow is lost and thermal barrier heat exchanger cooling is maintained, it is recommended that pump operation be continued no longer than 24 hours, provided the previously identified parameters remain within allowable limits. If these parameters reach the maximum allowable values, the pump should be stopped.

The operator will be alerted to a loss of seal injection flow to the RCPs by the low flow alarm (and indication) located on the main control board. If the loss of seal injection flow results in an increase in number 1 seal leakage, the operator will be alerted of a high number 1 seal leak rate by the high flow alarm (and flow recording) located on the main control board. The pump bearing area temperature and seal area temperature indications are monitored by the computer; the operator will be alerted of resultant high temperature in either area of the pump by CCW return high temperature alarms annunciated on the main control board and high bearing temperature alarms on the plant computer.

The excess letdown path is provided as an alternate letdown path from the RCS in the event that the normal letdown path is inoperable. Reactor coolant can be discharged from a cold leg to flow through the tube side of the excess letdown heat exchanger where it is cooled by component cooling water. Downstream of the heat exchanger a remote-manual control valve controls the letdown flow. The flow normally joins the number 1 seal discharge manifold and passes through the seal water return filter and heat exchanger to the suction side of the charging pumps. The excess letdown flow can also be directed to the reactor coolant drain tank or directly into the volume control tank via a spray nozzle. When the normal letdown line is not available, the normal purification path is also not in operation. Therefore, this alternate condition would allow continued power operation for a limited period of time, dependent on RCS chemistry and activity.

During plant heatup, the excess letdown flow path is placed in service and left in service to avoid an excessive thermal transient, which would occur if the system operation were initiated with a large differential temperature near the end of heatup. The excess letdown flow path is used to provide additional letdown capability during the final stages of plant heatup. This path removes some of the excess reactor coolant due to expansion of the system as a result of the RCS temperature increase.

Surges in RCS inventory due to load changes are accommodated for the most part in the pressurizer. The volume control tank provides surge capacity for reactor coolant expansion not accommodated by the pressurizer. The Volume Control Tank water level is controlled via a three way valve which directs letdown/makeup flow to the Volume Control Tank (“VCT” position)

or diverts it to the Boron Recycle System via the Recycle Hold-up Tank ("RHT" position). In addition, the three-way valve controller has an "Auto" position which controls the Volume Control Tank water level automatically by diverting all or a portion of the letdown flow to the Recycle Hold-up Tank when the Volume Control Tank exceeds its normal operating water level. The three-way valve is normally maintained in the "VCT" position with the control room operator manually controlling the Volume Control Tank water level. If the Volume Control Tank water level exceeds a preset maximum, an alarm is actuated and the letdown flow is automatically diverted to the Boron Recycle System. Additional alarms have been added to; 1) alert the control room operators when the three-way valve controller is not in the "VCT" position, and 2) provide early indication of an increasing Volume Control Tank water level ("high" water level alarm). These alarms were added to aid the control room operators detection of a potential inadvertent boron dilution event in Modes 3, 4 and 5. See Section 15.4.6 for additional information.

Low level in the volume control tank initiates makeup from the Reactor Makeup Control System. If the Reactor Makeup Control System does not supply sufficient makeup to keep the volume control tank level from falling to a lower level, a low alarm is actuated. Manual action may correct the situation or, if the level continues to decrease, a low-low level signal from both level channels causes the suction of the charging pumps to be transferred to the refueling water storage tank.

The reciprocating charging pump is also used to perform hydrostatic tests which verify the integrity and leak-tightness of the RCS. The pump can pressurize the RCS to the maximum designated test pressure. The hydrostatic test is performed prior to initial operation and is part of the periodic RCS inservice inspection program.

9.3.4.1.2.2 Reactor Coolant Purification and Chemistry Control System

Reactor coolant water chemistry specifications are given in Table 5.2-5.

pH Control

The pH control chemical employed is lithium hydroxide. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/inconel systems. In addition, lithium-7 is produced in the core region due to irradiation of the dissolved boron in the coolant.

The concentration of lithium-7 is maintained as required to support the hot full power pH_t range as specified in Table 5.2-5. If the concentration exceeds this range, as it may during the early stages of a core cycle, the cation bed demineralizer is employed in the letdown line in series operation with a mixed bed demineralizer. Since the amount of lithium to be removed is small and its buildup can be readily calculated, the flow through the cation bed demineralizer is not required to be full letdown flow. If the concentration of lithium-7 is below the specified limits, lithium hydroxide can be introduced into the RCS via the charging flow. The solution is prepared in the laboratory and poured into the chemical mixing tank. Reactor makeup water is then used to flush the solution to the suction manifold of the charging pumps.

Oxygen Control

Hydrazine is employed as an oxygen scavenging agent during reactor startup from cold condition or during reactor shutdown to a cold condition. Hydrazine addition is required when hydrogen is

not effective or limited driving force is present to combine oxygen and hydrogen. The hydrazine solution is introduced into the RCS in the same manner as described above for the pH control agent.

Dissolved hydrogen is employed to control and scavenge oxygen produced due to radiolysis of water in the core region. Sufficient partial pressure of hydrogen is maintained in the volume control tank such that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant. A pressure control maintains a minimum pressure in the vapor space of the volume control tank. This valve can be adjusted to provide the correct equilibrium hydrogen concentration (25 to 50 cc hydrogen at STP per kilogram of water). Hydrogen is supplied from the hydrogen bulk gas storage system.

Reactor Coolant Purification

Mixed bed demineralizers are provided in the letdown line to provide cleanup of the letdown flow. The demineralizers remove ionic corrosion products and certain fission products. During normal power operation one demineralizer is in continuous service and can be supplemented intermittently by the cation bed demineralizer, if necessary, for additional purification. The cation resin removes principally cesium and lithium isotopes from the purification flow. The second mixed bed demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

A bypass line and isolation valve also provide additional flexibility for cleanup of letdown flow. The line and valve allow letdown flow to be aligned such that the mixed bed demineralizers are bypassed and flow is directly to the cation bed demineralizer for cleanup. Letdown can also be aligned to flow to the BTRS demineralizer(s) for cleanup, either directly or in series with either a mixed bed or the cation bed demineralizer.

In Mode 3 below 500 degrees F, during RCS cleanup for outages, purification flow can be increased by opening all three letdown orifices.

A further cleanup feature is provided for use during Residual Heat Removal System operation. A RHR letdown flow control valve admits flow into the CVCS downstream of the containment isolation valves. Both demineralizers can be placed in service to allow for a more rapid cleanup.

Filters are provided at various locations to ensure filtration of particulate and resin fines and to protect the seals on the reactor coolant pumps.

Fission gases are removed from the reactor coolant by purging of the volume control tank to the Gaseous Waste Processing System.

9.3.4.1.2.3 Reactor Makeup Control System

The soluble neutron absorber (boric acid) concentration is controlled by the Reactor Makeup Control System. The Boron Thermal Regeneration System, if properly configured, may also be used. The Reactor Makeup Control System is also used to maintain proper reactor coolant inventory. In addition, for emergency boration and makeup, the capability exists to provide refueling water or 4 weight percent boric acid directly to the suction of the charging pump.

The Reactor Makeup Control System provides a manually pre-selected makeup composition to the charging pump suction header or to the volume control tank. The makeup control functions are those of maintaining desired operating fluid inventory in the volume control tank and adjusting reactor coolant boron concentration for reactivity control. Reactor makeup water and boric acid solution (4 weight percent) are blended together at the reactor coolant boron concentration for use as makeup to maintain volume control tank inventory or they can be used separately to change the reactor coolant boron concentration.

The boric acid is stored in two boric acid tanks. Two boric acid transfer pumps are provided with one pump normally required to provide boric acid to the suction header of the charging pumps, and the second pump in reserve. They are both aligned to take suction from separate boric acid tanks. On a demand signal by the reactor makeup controller, the pump starts and delivers boric acid to the suction header of the charging pumps. The pump can also be used to recirculate the boric acid tank fluid.

All portions of the CVCS which normally contain concentrated boric acid solution (4 weight percent boric acid) are required to be located with a heated area in order to maintain solution temperature at ≥65°F. If a portion of the system which normally contains concentrated boric acid solution is not located in a heated area, it must be provided with some other means (e.g., heat tracing) to maintain solution temperature at ≥65°F. Heat tracing is utilized inside the plant on 4% boric acid solution lines where potential freezing poses a recrystalization problem.

The reactor makeup water pumps take suction from the reactor make-up water storage tank and are employed for various makeup and flushing operations throughout the systems. There is one pump for each unit plus a common pump for both units. One of these pumps normally operates continuously and provides flow to the suction header of the charging pumps or the volume control tank through the letdown line and spray nozzle on demand.

During reactor operation, changes are made in the reactor coolant boron concentration for the following conditions:

1. Reactor startup - boron concentration must be decreased from shutdown concentration to achieve criticality.

2. Load follow - boron concentration must be either increased or decreased to compensate for the xenon transient following a change in load.

3. Fuel burnup - boron concentration must be decreased to compensate for fuel burnup and the buildup of fission products in the fuel.

4. Cold shutdown - boron concentration must be increased to the cold shutdown concentration.

5. Burnable Poiso depletion - dependent on the core design and time in core life, boron concentration may need to be increased to compensate for the depletion of burnable poisons in the fuel.

The Boron Thermal Regeneration System may be used to control boron concentration to compensate for xenon transients during load follow operations. Boron thermal regeneration can also be used in conjunction with dilution operations of the Reactor Makeup Control System to

reduce the amount of effluent to be processed by the Liquid Waste System or the Boron Recycle System.

The Reactor Makeup Control System can be set up for the following modes of operation:

1. Automatic makeup

   The “automatic makeup” mode of operation of the Reactor Makeup Control System provides blended boric acid solution, preset to match the boron concentration in the RCS. Automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the reactor coolant boron concentration.

   Under normal plant operating conditions, the mode selector switch is set in the “automatic makeup” position. This switch position establishes a preset control signal to the total makeup flow controller and establishes positions for the makeup stop valves for automatic makeup. The boric acid flow controller is set to blend to the same concentration of borated water as contained in the RCS. A preset low level signal from the volume control tank level controller causes the automatic makeup control action to start a boric acid transfer pump, open the makeup stop valve to the charging pump suction, and position the boric acid flow control valve and the reactor makeup water flow control valve. 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 point, the makeup is stopped. This operation may be terminated manually at any time.

   If the automatic makeup fails or is not aligned for operation and the tank level continues to decrease, a low level alarm is actuated. Manual action may correct the situation or, if the level continues to decrease, a low-low level signal opens the stop valves in the refueling water supply line to the charging pumps, and closes the stop valves in the volume control tank outlet line.

2. Dilution

   The “dilute” mode of operation permits the addition of a preselected quantity of reactor makeup water at a preselected flow rate to the RCS. The operator sets the mode selector switch to “dilute,” the total makeup flow controller setpoint to the desired flow rate, the total makeup batch integrator to the desired quantity and initiates system start. This opens the reactor makeup water flow control valve and opens the makeup stop valve to the volume control tank inlet. Excessive rise of the volume control tank water level is prevented by manual or automatic actuation of a three-way diversion valve which routes the reactor coolant letdown flow to the Boron Recycle System. When the preset quantity of water has been added, the batch integrator causes makeup to stop. The operation may be terminated manually at any time.

   Dilution can also be accomplished by operating the Boron Thermal Regeneration System in the boron storage mode.

3. Alternate dilution

The “alternate dilute” mode of operation is similar to the dilute mode except a portion of the dilution water flows directly to the charging pump suction and a portion flows into the volume control tank via the spray nozzle and then flows to the charging pump suction. This decreases the delay in diluting the RCS caused by directing dilution water to the volume control tank.

4. 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 RCS. The operator sets the mode selection switch to “borate,” the concentrated boric acid flow controller setpoint to the desired flow rate, the concentrated boric acid batch integrator to the desired quantity, and initiates system start. This opens the makeup stop valve to the charging pumps suction, positions the boric acid flow control valve, and starts the selected boric acid transfer pump, which delivers a 4 weight percent boric acid solution to the charging pumps 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 is added, the batch integrator causes makeup to stop. The operation may be terminated manually at any time.

Boration can also be accomplished by operating the Boron Thermal Regeneration System in the boron release mode. See Section 9.3.4.1.1.1.

5. Manual

The “manual” mode of operation permits the addition of a preselected quantity and blend of boric acid solution to the charging pump suction and volume control tank inlet, to the refueling water storage tank, to the recycle holdup tanks in the Boron Recycle System, or to some other location via a temporary connection. While in the manual mode of operation, automatic makeup to the RCS is precluded. The discharge flow path must be prepared by opening manual valves in the desired path.

The operator sets the mode selector switch to “manual,” the boric acid and total makeup flow controllers to the desired flow rates, the boric acid and total makeup batch integrators to the desired quantities, and actuates the makeup start switch.

The start switch actuates the boric acid flow control valve and the reactor makeup water flow control valve and starts the boric acid transfer pump.

When the preset quantities of boric acid and reactor makeup water have been added, the batch integrators cause makeup to stop. This operation may be stopped manually by actuating the makeup stop switch.

If either batch integrator is satisfied before the other has recorded its required total, the pump and valve associated with the integrator which has been satisfied will terminate flow. The flow controlled by the other integrator will continue until that integrator is satisfied. In the manual mode, the boric acid flow is terminated first to prevent piping systems from remaining filled with 4 weight percent boric acid solution.

The quantities of boric acid and reactor makeup water injected are totalized by the batch counters and the flow rates are recorded on strip recorders. Deviation alarms sound for both boric acid and reactor makeup water if flow rates deviate from setpoints.

9.3.4.1.2.4 Boron Thermal Regeneration System

Comanche Peak currently does not utilize load follow operation. Some of the design capabilities discussed below are therefore not currently utilized. See Section 9.3.4.1.1.1.

Downstream of the mixed bed demineralizers, the letdown flow can be diverted to the Boron Thermal Regeneration System where part or all of the letdown flow can be treated when boron concentration changes are desired.

Additionally, the BTRS demineralizer(s) can be utilized to provide for supplemental or enhanced cleanup of letdown flow. After processing, the flow is returned to a point upstream of the reactor coolant filter.

Storage and release of boron during load follow operation is determined by the temperature of fluid entering the thermal regeneration demineralizers. A chiller unit and a group of heat exchangers are employed to provide the desired fluid temperatures at the demineralizer inlets for either storage or release operation of the system.

The flow path through the Boron Thermal Regeneration System is different for the boron storage and the boron release operations. During boron storage, the letdown stream enters the moderating heat exchanger and from there it passes through the letdown chiller heat exchanger. These two heat exchangers cool the letdown stream prior to its entering the demineralizers. The letdown reheat heat exchanger is valved out on the tube side and performs no function during boron storage operations. The temperature of the letdown stream at the point of entry to the demineralizers is controlled automatically by the temperature control valve which controls the shell side flow to the letdown chiller heat exchanger. After passing through the demineralizers, the letdown enters the moderating heat exchanger shell side, where it is heated by the incoming letdown stream before going to the volume control tank.

Therefore, for boron storage, a decrease in the boric acid concentration in the reactor coolant is accomplished by sending the letdown flow at relatively low temperatures to the thermal regeneration demineralizers. The resin, which was depleted of boron at high temperature during a prior boron release operation, is now capable of storing boron from the low temperature letdown stream. Reactor coolant with a decreased concentration of boric acid leaves the demineralizers and is directed to the RCS via the charging system.

During the boron release operation, the letdown stream enters the moderating heat exchanger tube side, bypasses the letdown chiller heat exchanger, and passes through the shell side of the letdown reheat heat exchanger. The moderating and letdown reheat heat exchangers heat the letdown stream prior to its entering the resin beds. The temperature of the letdown at the point of entry to the demineralizers is controlled automatically by the temperature control valve which controls the flow rate on the tube side of the letdown reheat heat exchanger. After passing through the demineralizers, the letdown stream enters the shell side of the moderating heat exchanger, passes through the tube side of the letdown chiller heat exchanger and then goes to the volume control tank. The temperature of the letdown stream entering the volume control tank is controlled automatically by adjusting the shell side flow rate on the letdown chiller heat

exchanger. Thus, for boron release, an increase in the boric acid concentration in the reactor coolant is accomplished by sending the letdown flow at relatively high temperatures to the thermal regeneration demineralizers. The water flowing through the demineralizers now releases boron which was stored by the resin at low temperature during a previous boron storage operation. The boron enriched reactor coolant is returned to the RCS via the charging system.

Although the Boron Thermal Regeneration System is primarily designed to compensate for xenon transients occurring during load follow, it can also be used to handle boron swings far in excess of the design capacity of the demineralizers. If used for startup dilution, for example, the resin beds are first saturated, then washed off to the Boron Recycle System, then again saturated and washed off. This operation continues until the desired dilution in the RCS is obtained.

A thermal regeneration demineralizer can also be used as a deborating demineralizer, which would be used to dilute the RCS down to very low boron concentrations towards the end of a core cycle. To make such a bed effective, the effluent concentration from the bed must be kept very low, close to zero ppm boron. This low effluent concentration can be achieved by using fresh resin. Use of fresh resin can be coupled with the normal replacement cycle of the resin; one resin bed being replaced during each core cycle.

As an additional function, the thermal regeneration (BTRS) demineralizer(s) may be loaded with various types of resin in order to provide supplemental or enhanced cleanup of letdown flow. This may be accomplished by aligning letdown such that flow will be directly to the BTRS demineralizer(s) or by placing these demineralizer(s) in series with either a mixed bed or the cation bed demineralizer.

9.3.4.1.2.5 Components Description

A summary of principal component design parameters is given in Table 9.3-7, and safety classifications and design codes are given in Section 3.2.

All CVCS piping that handles 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.

Charging Pumps

Three charging pumps are supplied to inject coolant into the RCS. Two of the pumps are of the single speed, horizontal, centrifugal type and the third is a positive displacement (reciprocating) pump equipped with variable speed drive. All parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other material of adequate corrosion resistance. The centrifugal pump seals and the reciprocating pump stuffing box are provided with leakoffs to collect the leakage before it can leak to the atmosphere. The reciprocating pump design prevents lubricating oil from contaminating the charging flow. There is a minimum flow recirculation line to protect the centrifugal charging pumps from a closed discharge valve condition.

Charging flow rate is determined from a pressurizer level signal. The means of flow control for the reciprocating pump is by variation of pump speed. The reciprocating charging pump is also used to hydrotest the RCS. When operating a centrifugal charging pump, the flow paths remain

the same but charging flow control is accomplished by a modulating valve on the discharge side of the centrifugal pumps. The centrifugal charging pumps also serve as high head safety injection pumps in the Emergency Core Cooling System. A description of the charging pump function upon receipt of a safety injection signal is given in Section 6.3.2.2.5.

Boric Acid Transfer Pumps

Two canned motor pumps are supplied per unit. One pump is normally required to supply boric acid to the suction header of the charging pumps while the second serves as a standby. Manual or automatic initiation of the Reactor Coolant Makeup System will start the one pump to provide normal makeup of boric acid solution to the suction header of the charging pumps. Miniflow from this pump flows back to the associated boric acid tank and helps maintain thermal equilibrium. The recirculation of the boric acid tanks using the boric acid transfer pump allows the location of a sample point at the boric acid transfer pump discharge. For a remote sampling station, purging of sample lines ensures a representative sample will be taken. The standby pump is also aligned and can be used intermittently to circulate boric acid solution through the other tank to maintain thermal equilibrium in this part of the system. Emergency boration, supplying 4 weight percent boric acid solution directly to the suction of the charging pumps, can be accomplished by manually starting either or both pumps. The transfer pumps may also function to transfer boric acid solution from the batching tank to the boric acid tanks. All parts in contact with the solution are of austenitic stainless steel.

Chiller Pumps

Two centrifugal pumps circulate the water through the chilled water loop in the Boron Thermal Regeneration System. One pump is normally operated, with the second serving as a standby.

Because load follow operation is currently not utilized, the Chiller Pumps are normally removed from service, the chiller loops having been drained. See Section 9.3.4.1.1.1.

Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover heat from the letdown flow by reheating the charging flow, which reduces thermal effects on the charging penetrations into the reactor coolant loop piping.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flow through the tubes. The unit is constructed of austenitic stainless steel, and is of all welded construction.

The temperatures of both outlet streams from the heat exchanger are monitored with indication given in the Control Room. A high temperature alarm is actuated on the main control board if the temperature of the letdown stream exceeds desired limits.

Letdown Heat Exchanger

The letdown heat exchanger cools the letdown stream to the operating temperature of the downstream demineralizers. Reactor coolant flows through the tube side of the exchanger while component cooling water flows through the shell side. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel.

The low pressure letdown valve, located downstream of the heat exchanger, maintains the pressure of the letdown flow upstream of the heat exchanger in a range sufficiently high to prevent two phase flow. Pressure indication and high pressure alarm are provided on the main control board.

The letdown temperature controller indicates and controls the temperature of the letdown flow exiting from the letdown heat exchanger. A temperature sensor, which is part of the CVCS, provides input to the controller in the Component Cooling Water System. The exit temperature of the letdown stream is thus controlled by regulating the component cooling water flow through the letdown heat exchanger. Temperature indication is provided on the main control board. If the outlet temperature from the heat exchanger is excessive, a high temperature alarm is actuated and a temperature controlled valve diverts the letdown directly to the volume control tank.

The outlet temperature from the shell side of the heat exchanger is allowed to vary over an acceptable range compatible with the equipment design parameters and required performance of the heat exchanger in reducing letdown stream temperature.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow. This letdown flow rate is equivalent to the portion of the nominal seal injection flow which flows into the RCS through the reactor coolant pump labyrinth seals.

The excess letdown heat exchanger can be employed either when normal letdown is temporarily out of service to maintain the reactor in operation or it can be used to supplement maximum letdown during the final stages of heatup. The letdown flows through the tube side of the unit and component cooling water is circulated through the shell. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded.

A temperature detector measures the temperature of the excess letdown flow downstream of the excess letdown heat exchanger. Temperature indication and high temperature alarm are provided on the main control board.

A pressure sensor indicates the pressure of the excess letdown flow downstream of the excess letdown heat exchanger and excess letdown control valve. Pressure indication is provided on the main control board.

Seal Water Heat Exchanger

The seal water heat exchanger is designed to cool fluid from three sources: reactor coolant pump number 1 seal leakage, reactor coolant discharged from the excess letdown heat exchanger, and miniflow from a centrifugal charging pump. Reactor coolant flows through the tube side of the heat exchanger and component cooling water is circulated through the tubes. The design flow rate through the tube side is equal to the sum of the nominal excess letdown flow, maximum design reactor coolant pump seal leakage, and miniflow from one centrifugal charging pump. The unit is designed to cool the above flow to the temperature normally maintained in the volume control tank. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

Moderating Heat Exchanger

The moderating heat exchanger operates as a regenerative heat exchanger between incoming and outgoing streams to and from the thermal regeneration demineralizers.

The incoming letdown flow enters the tube side of the moderating heat exchanger. The shell side fluid, which comes directly from the thermal regeneration demineralizers, enters at low temperature during boron storage and high temperature during boron release.

Letdown Chiller Heat Exchanger

During the boron storage operation, the process stream enters the tube side of the letdown chiller heat exchanger after leaving the tube side of the moderating heat exchanger. The letdown chiller heat exchanger cools the process stream to allow the thermal regeneration demineralizers to remove boron from the coolant. The desired cooling capacity is adjusted by controlling the chilled water flow rate passed through the shell side of the heat exchanger.

The letdown chiller heat exchanger is also used during the boron release operation to cool the liquid leaving the thermal regeneration demineralizers to ensure that its temperature does not exceed that of normal letdown to the volume control tank.

Because load follow operation is currently not utilized, the Chiller Pumps are normally removed from service, the chiller loops having been drained. See Section 9.3.4.1.1.1.

Letdown Reheat Heat Exchanger

The letdown reheat heat exchanger is used only during boron release operations and it is then used to heat the process stream. Water used for heating is diverted from the letdown line upstream of the letdown heat exchanger, passed through the tube side of the letdown reheat heat exchanger and then returned to the letdown stream upstream of the letdown heat exchanger.

Because load follow operation is currently not utilized, the Letdown Reheat Heat Exchanger tube side is normally isolated and drained. See Section 9.3.4.1.1.1.

Volume Control Tank

The volume control tank provides surge capacity for part of the reactor coolant expansion volume not accommodated by the pressurizer. When the level in the tank reaches a preset maximum level setpoint, the remainder of the expansion volume is accommodated by diversion of the letdown stream to the Boron Recycle System. The tank also provides a means for introducing hydrogen into the coolant to maintain the required equilibrium concentration of 25 to 50 cc hydrogen (at STP) per kilogram of water and is used for degassing the reactor coolant. It also serves as a head tank for the charging pumps.

A spray nozzle located inside the tank on the letdown line provides liquid to gas contact between the incoming fluid and the hydrogen atmosphere in the tank.

Hydrogen (from the hydrogen bulk gas storage system) is supplied to the volume control tank while a remotely operated vent valve, discharging to the Gaseous Waste Processing System,

permits continuous removal of gaseous fission products which are stripped from the reactor coolant and collected in this tank. Relief protection, gas space sampling, and nitrogen purge connections are also provided. The tank can also accept the seal water return flow from the reactor coolant pumps although this flow normally goes directly to the suction of the charging pumps.

Argon gas is intermittently injected to the Volume Control Tank to increase radioactivity in the RCS. The Argon gas is injected through a port located in the Process Sampling System and is equipped with over-pressure protection. The increase in radioactivity allows for early detection of low volume, primary to secondary leakage through the steam generators. The Argon concentration should be increased based on the radiological needs for detection and shall remain within the specified design parameters and technical specifications.

Volume control tank pressure is monitored with indication given in the Control Room. An Alarm is actuated in the Control Room for high and low pressure conditions. The volume control tank pressure control valve is automatically closed by the low pressure signal.

Two level channels govern the water inventory in the volume control tank.

If the volume control tank level rises above the normal operating range and the three-way valve is in the “Auto” position, one level channel provides an analog signal to the proportional controller which modulates the three-way valve downstream of the reactor coolant filter to maintain the volume control tank level within the normal operating band. The three-way valve can split letdown flow so that a portion goes to the Boron Recycle System and a portion to the volume control tank. The controller would operate in this fashion during a dilution operation when reactor makeup water is being fed to the volume control tank from the Reactor Makeup Control System.

If the modulating function of the channel fails and the volume control tank level continues to rise, a level alarm will alert the operator to the malfunction. Letdown flow will then be diverted.

During normal power operation, a low level in the volume control tank initiates auto makeup which injects a pre-selected blend of boric acid solution and reactor makeup water into the charging pump suction header. When the volume control tank level is restored to normal, auto makeup stops.

If the automatic makeup fails or is not aligned for operation and the tank level continues to decrease, a low level alarm is actuated.

Manual action may correct the situation or, if the level continues to decrease, a low-low signal from both level channels opens the stop valves in the refueling water supply line, and closes the stop valves in the volume control tank outlet line.

Boric Acid Tanks

The combined boric acid tank capacity is sized to store sufficient boric acid solution for refueling plus enough for a cold shutdown from full power operation immediately following refueling with the most reactive control rod not inserted. The two boric acid tanks are shared between Units 1 and 2. Each tank has a nitrogen sparger installed inside the tank. Nitrogen from the plant bulk low pressure nitrogen gas system is injected into the boric acid solution to reduce and maintain low dissolved oxygen levels within the fluid.

The concentration of boric acid solution in storage is maintained between 4 and 4.4 percent by weight. Periodic manual sampling and corrective action, if necessary, assure that these limits are maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the boron concentration.

A temperature sensor provides temperature measurement of each tank's contents. Temperature indication as well as high and low temperature alarms are provided on the main control board.

Two level detectors indicate the level in each boric acid tank. Level indication is provided on the Unit 1 Main Control Board, with high, low, low-low and empty level alarms provided on both the Unit 1 and Unit 2 Main Control Boards. The high alarm indicates that the tank may soon overflow. The low alarm warns the operator to start makeup to the tank. The low-low alarm is set to indicate the minimum level of boric acid in the tank to ensure sufficient boric acid is available for a cold shutdown with one stuck rod. The empty level alarm is set to give warning of loss of pump suction.

Batching Tank

The batching tank is used for mixing a makeup supply of boric acid solution for transfer to the boric acid tanks. One batching tank is shared between Units 1 and 2.

A local sampling point is provided for verifying the solution concentration prior to transferring it out of the tank. The tank is provided with an agitator to improve mixing during batching operations (and thereby provide a representative sample), and a steam jacket for heating the boric acid solution.

Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of caustic solutions for pH control and hydrazine solution for oxygen scavenging.

Chiller Surge Tank

The chiller surge tank handles the thermal expansion and contraction of the water in the chiller loop. The surge volume in the tank also acts as a thermal buffer for the chiller. The fluid level in the tank is monitored with level indication and high and low level alarms provided on the main control board. Due to current operating philosophy, the low level alarm is no longer in service.

Because load follow operation is currently not utilized, the Chiller Surge Tank is normally drained. A high level alarm is required to be in service when the BTRS is in service. The high level alarm provides an alert if water leaks from the shell side to the tube side of the Letdown Chiller Heat Exchanger and fills the Chiller Surge Tank. A low level alarm is required when the BTRS chiller loop is filled and used to provide chilled water to the Letdown Chiller Heat Exchanger. However, the chilled water loop is normally drained because load follow operation is currently not utilized. See Section 9.3.4.1.1.1.

Discharge Dampener Tank

The discharge dampener is a spherical pressure vessel which uses centrifugal flow patterns to dampen pressure pulsations at the discharge of the positive displacement pump.

Suction Stabilizer Tank

The suction stabilizer is a tank designed to reduce fluid pulsations and acceleration head loss. A cover gas provides pulsation absorption capacity. To prevent excessive gas from accumulating, a vent path to the VCT is provided. To make up for any gas absorbed by the water, a gas makeup system is used. When the stabilizer water level increases, a level signal opens a valve to supply gas. The Operator then momentarily opens the two down stream isolation valves. When the level drops to a predetermined level, the supply valve closes. A hi-hi alarm on the main control board is also provided to warn that the stabilizer is approaching a water solid condition. The gas makeup source will be pre-aligned for consistency with the VCT cover gas. A pressure regulator is located in the gas makeup line to restrict flow to prevent perturbations of the stabilizer water level during gas addition. The vent and gas makeup paths are automatically isolated on an “S” signal.

Mixed Bed Demineralizers

Two flushable mixed bed demineralizers assist in maintaining reactor coolant purity. A lithium-form cation resin and hydroxyl-form anion resin may be charged into the demineralizers. The anion resin is converted to the borate form in operation. Both types of resin remove fission and corrosion products. 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 capable of removing ionic impurities for one core cycle with one percent of the rated core thermal power being generated by defective fuel rods. If the mixed bed demineralizers are in use, one demineralizer is normally in service with the other in standby.

A temperature sensor monitors the temperature of the letdown flow down stream of the letdown heat exchanger and if the letdown temperature exceeds the maximum allowable resin operating temperature (approximately 140°F), a three-way valve is automatically actuated to bypass the flow around the demineralizers. Temperature indication and high alarm are provided on the main control board. The air operated three-way valve failure mode directs flow to the volume control tank.

Cation Bed Demineralizers

A flushable demineralizer with cation resin in the hydrogen form may be used intermittently to control the concentration of lithium-7 which builds up in the coolant from the boron-10 (n,α) lithium-7 reaction. The demineralizer also is capable of maintaining the cesium-137 concentration in the coolant below 1.0 μCi/cc with 1 percent defective fuel. The resin bed is designed to reduce the concentration of ionic isotopes, particularly cesium, yttrium, and molybdenum by a minimum factor of 10.

The demineralizer is capable of removing ionic impurities for one core cycle with 1 percent of the rated core thermal power being generated by defective fuel rods.

Thermal Regeneration Demineralizers

The function of the thermal regeneration demineralizers is to store the total amount of boron that must be removed from the RCS to accomplish the required dilution during a load cycle in order to

compensate for xenon buildup resulting from a decreased power level. Furthermore, the demineralizers must be able to release the previously stored boron to accomplish the required boration of the reactor coolant during the load cycle in order to compensate for a decrease in xenon concentration resulting from an increased power level.

As an additional function, the thermal regeneration (BTRS) demineralizer(s) may be loaded with various types of resin in order to provide supplemental or enhanced cleanup of letdown flow. This may be accomplished by aligning letdown such that flow will be directly to the BTRS demineralizer(s) or by placing these demineralizer(s) in series with either a mixed bed or the cation bed demineralizer.

The thermally reversible ion storage capacity of the resin applies only to borate ions. The capacity of the resin to store other ions is not thermally reversible. Thus, during boration, when borate ions are released by the resin, there is no corresponding release of the ionic fission and corrosion products stored on the resin.

The thermal regeneration demineralizer resin capacity is directly proportional to the solution boron concentration and inversely proportional to the temperature. Further, the differences in capacity as a function of both boron concentration and temperature are reversible. For the 50 to 140°F temperature cycle this reversible capacity varies from the beginning of a core cycle to the end of core life by a factor of about 2.

The demineralizers are of the type that can accept flow in either direction. The flow direction during boron storage is therefore always opposite to that during release. This provides much faster response when the beds are switched from storage to release and vice versa, than would be the case if the demineralizers could accept flow in only one direction.

Temperature instrumentation is provided upstream of the thermal regeneration demineralizers to control the temperature of the process flow. During boron storage operations, it controls the flow through the shell side of the letdown chiller heat exchanger to maintain the process flow at 50°F as it enters the demineralizers. During boron release operations, it controls the flow through the tube side of the letdown reheat heat exchanger to maintain the process flow at 140°F as it enters the demineralizers. Temperature indication and a high temperature alarm are provided on the main control board.

An additional temperature instrument is provided to protect the demineralizer resins from a high temperature condition. On reaching the high temperature setpoint, an alarm is sounded on the main control board and the letdown flow is diverted to the volume control tank from a point upstream of the mixed bed demineralizers.

Failure of the temperature controls resulting in hot water flow to the demineralizers would result in a release of boron stored on the resin with a resulting increase in reactor coolant boron concentration and increased margin for shutdown. If the temperature of the resin rises significantly above 140°F, the number of ion storage will gradually decrease, thus reducing the capability of the resin to remove boron from the process stream. Degradation of ion removal capability will occur for temperatures of approximately 160°F and above. The extent of the degradation and rate at which it will occur depend upon the temperature experienced by the resin and the length of time that the resin experiences this elevated temperature.

Failure of the temperature control system resulting in cold water flow to the demineralizers would result in storage of boron on the resin and reduction of the reactor coolant boron concentration. The amount of reduction in reactor coolant boron concentration is limited by the capacity of the resin to remove boron from the water. As the boron concentration is reduced, the control rods would be driven into the core to maintain power level. If the rods were to reach the rod insertion limit setpoint, an alarm would be actuated informing the operator that the shutdown margin should be verified within limits or to take actions to restore the shutdown margin in accordance with Technical Specifications.

Reactor Coolant Filter

The reactor coolant filter is located in the letdown line upstream of the volume control tank. The filter collects resin fines and particulates from the letdown stream. The nominal flow capacity of the filter is greater than the maximum purification flow rate.

Two local pressure indicators are provided to show the pressures upstream and downstream of the reactor coolant filter and thus provide filter differential pressure.

Seal Water Injection Filters

Two seal water injection filters are located in parallel in a common line to the reactor coolant pump seals; they collect particulate matter that could be harmful to the seal faces. Each filter is sized to accept flow in excess of the normal seal water flow requirements.

A differential pressure indicator monitors the pressure drop across each seal water injection filter and gives local indication with high differential pressure alarm on the main control board.

Seal Water Return Filter

This filter collects particulates from the reactor coolant pump seal water return and from the excess letdown flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from all reactor coolant pumps.

Two local pressure indicators are provided to show the pressures upstream and downstream of the filter and thus provide differential pressure across the filter.

Boric Acid Filter

The boric acid filter collects particulates from the boric acid solution being pumped from the boric acid tanks by the boric acid transfer pumps. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously.

Local pressure indicators indicate the pressure upstream and downstream of the boric acid filter and thus provide filter differential pressure.

Letdown Orifices

Three letdown orifices are provided to reduce the letdown pressure from reactor conditions and to control the flow of reactor coolant leaving the RCS. The orifices are placed into or out of service by remote operation of their respective isolation valves. One orifice is designed for

normal letdown flow with the other two serving as standby. One or both of the standby orifices may be used in parallel with the normally operating orifice for either flow control when the RCS pressure is less than normal or greater letdown flow during maximum purification or heatup. Each orifice consists of an assembly which provides for permanent pressure loss without recovery, and is made of austenitic stainless steel or other adequate corrosion resistant material.

A flow monitor provides indication in the Control Room of the letdown flow rate, and a high alarm to indicate unusually high flow.

A low pressure letdown controller located downstream of the letdown heat exchanger controls the pressure upstream of the letdown heat exchanger to prevent flashing of the letdown liquid. Pressure indication and high pressure alarm are provided on the main control board.

Number 1 Seal Bypass Orifice

An orifice in each reactor coolant pump number 1 seal bypass line is only in service during startup or shutdown when the RCS pressure is low. The bypass flow is necessary to ensure adequate flow for cooling of the pump's lower radial bearing and to limit the temperature rise of the water cooling the number 1 seal. The orifice is constructed of austenitic stainless steel and designed to pass adequate flow for the differential pressure existing at the lowest allowable RCS pressure for reactor coolant pump operation.

Chiller

Because load follow operation is currently not utilized, the Chiller is normally removed from service, the chiller loops having been drained. See Section 9.3.4.1.1.1.

The purpose of the chiller is two fold:

1. To cooldown the process stream during storage of boron on the resin.

2. To maintain an outlet temperature from the Boron Thermal Regeneration System at or below 115°F during release of boron.

Valves

Where pressure and temperature conditions permit, diaphragm type valves are used to essentially eliminate leakage to the atmosphere. All packed valves which are larger than 2 inches and which are designated for radioactive services are provided with a stuffing box and lantern leakoff connections. All control (modulating) and three-way valves are either provided with stuffing box and leakoff connections or are totally enclosed. Leakage to the atmosphere is essentially zero for these valves. Basic material of construction is stainless steel for all valves which handle radioactive liquid or boric acid solutions.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction.

1. Charging line downstream of regenerative heat exchanger

If the charging side of the regenerative heat exchanger is isolated while the hot letdown flow continues at its maximum rate, the volumetric expansion of coolant on the charging side of the heat exchanger is relieved to the RCS through a spring loaded check valve.

2. Letdown line downstream of letdown orifices

The pressure relief valve downstream of the letdown orifices protects the low pressure piping and the letdown heat exchanger from overpressure when the low pressure piping is isolated. The capacity of the relief valve exceeds the maximum flow rate through all letdown orifices. The valve set pressure is equal to the design pressure of the letdown heat exchanger tube side.

3. Letdown line downstream of low pressure letdown valve

The pressure relief valve downstream of the low pressure letdown valve protects the low pressure piping and equipment from overpressure when this section of the system is isolated. The overpressure may result from leakage through the low pressure letdown valve. The capacity of the relief valve exceeds the maximum flow rate through all letdown orifices. The valve set pressure is equal to the design pressure of the demineralizers.

4. Volume control tank

The relief valve on the volume control tank permits the tank to be designed for a lower pressure than the upstream equipment. This valve has a capacity greater than the summation of the following items: maximum letdown, normal seal water return, excess letdown and nominal flow from one reactor makeup water pump. The valve set pressure equals the design pressure of the volume control tank.

5. Charging pump suction

For Unit 1:

A relief valve on the charging pump suction header relieves pressure that may build up if the suction line isolation valves are closed or if the system is overpressurized. The valve set pressure is equal to the design pressure of the associated piping and equipment.

For Unit 2:

A relief valve on the charging pump suction header and relief valves installed on each Unit 2 charging pump suction line relieve pressure that may build up if the suction line isolation valves are closed or if the system is overpressurized. The valve set pressures are equal to the design pressure of the associated piping and equipment.

6. Seal water return line (inside Containment)

This relief valve is designed to relieve overpressurization in the seal water return piping inside the Containment if the motor operated isolation valve is closed. The valve is designed to relieve the total leakoff flow from the number 1 seals of the reactor coolant pumps plus the design excess letdown flow. The valve is set to relieve at the design pressure of the piping.

7. Seal water return line (charging pumps bypass flow)

   This relief valve protects the seal water heat exchanger and its associated piping from overpressurization. If either of the isolation valves for the heat exchanger are closed and if the bypass line is closed, the piping would be overpressurized by the miniflow from the centrifugal charging pumps. The valve is sized to handle the miniflow from the centrifugal charging pumps. The valve is set to relieve at the design pressure of the heat exchanger.

8. Positive displacement pump discharge

   The pressure relief valve on the positive displacement pump discharge line relieves the rated pumping capacity if the pump is started with the discharge isolation valve closed. The set pressure of the valve is equal to the design pressure of the pump discharge piping.

9. Letdown reheat heat exchanger

   The relief valve is located on the piping leading from the shell side of the heat exchanger. If the shell side were isolated while flow was maintained in the tube side, overpressurization could occur. The valve is set to relieve at the design pressure of the heat exchanger shell side.

   Because load follow operation is currently not utilized, the Letdown Reheat Heat Exchanger tube side is normally isolated and drained. Water that leaks into the drained tube side of the heat exchanger, whether leaking from the shell side into the tube side or leaking past the seats of the isolation valves, is drained into floor drains appropriate for water which potentially contains radioactivity. See Section 9.3.4.1.1.1.

10. Letdown chiller heat exchanger

    The relief valve is located on the piping leading from the shell side of the heat exchanger. If the shell side were isolated while flow was maintained in the tube side, overpressurization could occur. The valve is set to relieve at the design pressure of the heat exchanger shell side.

    Because load follow operation is currently not utilized, the Letdown Chiller Heat Exchanger shell side is normally drained. Water that leaks into the drained shell side of the heat exchanger from the tube side would be directed to the Chiller Surge Tank which has a high level alarm. See Section 9.3.4.1.1.1.

11. Steam line to batching tank

    The relief valve on the steam line to the batching tank protects the low pressure piping and batching tank heating jacket from overpressure when the condensate return line is isolated. The capacity of the relief valve equals the maximum expected steam inlet flow. The set pressure equals the design pressure of the heating jacket.

Piping

All CVCS piping that handles radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal of maintenance and hydrostatic testing.

9.3.4.1.2.6 System Operation

The following are typical operating methods for this system.

Reactor Startup

Reactor startup is defined as the operations which bring the reactor from cold shutdown to normal operating temperature and pressure.

It is assumed that:

1. Normal residual heat removal is in progress.

2. RCS boron concentration is at the cold shutdown concentration.

3. Reactor Makeup Control System is set to provide makeup at the cold shutdown concentration.

4. RCS is either water solid or drained to minimum level for the purpose of refueling or maintenance. If the RCS is water solid, system pressure is maintained by operation of a charging pump and controlled by the low pressure letdown valve in the letdown line (letdown is achieved via the RHRS).

5. The charging and letdown lines of the CVCS are filled with coolant at the cold shutdown boron concentration. The letdown orifice isolation valve are closed.

If the RCS requires filling and venting:

Two methods are available to fill the RCS. One method is to makeup to the RCS through CVCS via charging and letdown. The Reactor Coolant Pumps are sequentially operated and the system vented until all air is removed. The other method is by establishing a vacuum on the RCS and filling the system with makeup from CVCS with the charging pump suctions aligned preferably to the RWST.

1. One charging pump is started, which provides blended flow from the Reactor Makeup Control System at the cold shutdown boron concentration.

2. The vents on the head of the reactor vessel and pressurizer are opened.

3. The RCS is filled and the vents closed.

The system pressure is raised by using the charging pump and controlled by the low pressure letdown valve. When the system pressure is adequate for operation of the reactor coolant pumps, seal water flow to the pumps, is established and the pumps are operated and vented

sequentially until all gases are cleared from the system. Final venting takes place at the pressurizer.

Vacuum Fill:

Filling the Reactor Coolant System is started by establishing a vacuum on the system while controlling the RCS level with adjustment of CVCS pressure control. Once a vacuum has been established the charging pump suctions are aligned and charging flow adjusted. The reduced inventory instrumentation and pressurizer instrumentation are used to monitor the RCS level. The rate of filling the RCS is reduced as the Pressurizer begins to indicate a level increase. When the Pressurizer indicates sufficient level is present, the connections to the RCS are filled. After the connections are filled, the RCS fill is essentially completed, so vacuum is broken. Normal charging is established and the RCS is ready to continue being filled to a solid condition or to establish a pressurizer bubble.

After the filling and venting operations are completed, charging and letdown flows are established. All pressurizer heaters are energized and the reactor coolant pumps are employed to heat up the system. At this point, steam formation in the pressurizer is accomplished by manual control of the charging flow and automatic pressure control of the letdown flow. When the pressurizer water level reaches the no-load programmed setpoint, the pressurizer level control is shifted to control the charging flow to maintain programmed level. The RHRS is then isolated from the RCS and the normal letdown path is established. The pressurizer heaters are now used to increase RCS pressure.

The reactor coolant boron concentration is now reduced either by operating the Reactor Makeup Control System in the “dilute” mode or by operating the Boron Thermal Regeneration System in the boron storage mode and, when the resin beds are saturated, washing off the beds to the Boron Recycle System. The reactor coolant boron concentration is corrected to the point where the control rods may be withdrawn and criticality achieved. Nuclear heatup may then proceed with corresponding manual adjustment of the reactor coolant boron concentration to balance the temperature coefficient effects and maintain the control rods within their operating range. During heatup, the appropriate combination of letdown orifices is used to provide necessary letdown flow.

Prior to or during the heating process, the CVCS is employed to obtain the correct chemical properties in the RCS. The Reactor Makeup Control System is operated on a continuing basis to ensure correct control rod position. Chemicals are added through the chemical mixing tank as required to control reactor coolant chemistry such as pH and dissolved oxygen content. Hydrogen overpressure is established in the volume control tank to assure the appropriate hydrogen concentration in the reactor coolant.

Power Generation and Hot Standby Operation

1. Base Load

   At a constant power level, the rates of charging and letdown are dictated by the requirements for seal water to the reactor coolant pumps and the normal purification of the RCS. One charging pump is employed and charging flow is controlled automatically from pressurizer level. The only adjustments in boron concentration necessary are those to compensate for core burnup. These adjustments are made at infrequent intervals to

maintain the control groups within their allowable limits. Rapid variations in power demand are accommodated automatically by control rod movement. If variations in power level occur, and the new power level is sustained for long periods, some adjustment in boron concentration may be necessary to maintain the control groups within their maneuvering band.

During normal operation, normal letdown flow is maintained and demineralizer(s) are utilized to provide reactor coolant purification. Reactor coolant samples are taken periodically to check boron concentration, water quality, pH and activity level. The charging flow to the RCS is controlled automatically by the pressurizer level control signal through the discharge header flow control valve or the positive displacement pump speed controller.

2. Load Follow

Comanche Peak currently does not utilize load follow operation. See Section 9.3.4.1.1.1.

A power reduction will initially cause a xenon buildup followed by xenon decay to a new, lower equilibrium value. The reverse occurs if the power level increases; initially, the xenon level decreases and then it increases to a new and higher equilibrium value associated with the amount of the power level change.

The Boron Thermal Regeneration System can be used to vary the reactor coolant boron concentration to compensate for xenon transients occurring when reactor power level is changed. The Reactor Makeup Control System may also be used to vary the boron concentration in the reactor coolant.

The original design of the BTRS is to control the changes in the reactor coolant concentration to compensate for xenon transients during load follow operation without adding makeup for either boration or dilution. This is not currently utilized at Comanche Peak. The system is used to reduce reactor coolant boron concentration at the end of a core cycle using BTRS demineralizers loaded with anion resins. As a result, the BTRS chiller and associated components are normally removed from service, and the heating source for the reheat letdown heat exchanger is normally isolated. However, they are available to be used as desired.

The most important intelligence available to the plant operator, enabling him to determine whether dilution or boration of the RCS is necessary, is the position of the control rods. For example, if the control rods are below their desired position, the operator must borate the reactor coolant to bring the rods outward. If, on the other hand, the control rods are above their desired position, the operator must dilute the reactor coolant to bring the rods inward.

During periods of plant loading, the reactor coolant expands as its temperature rises. The pressurizer absorbs this expansion as the level controller raises the level setpoint to the increased level associated with the new power level. The excess coolant due to RCS expansion is letdown and stored in the volume control tank. During this period, the flow through the letdown orifice remains constant and the charging flow is reduced by the pressurizer level control signal, resulting in an increased temperature at the regenerative heat exchanger outlet. The temperature controller downstream from the letdown heat

exchanger increases the component cooling water flow to maintain the desired letdown temperature.

During periods of plant unloading, the charging flow is increased to make up for the coolant contraction not accommodated by the programmed reduction in pressurizer level.

3. Hot Standby

If required, for periods of maintenance, or following spurious reactor trips, the reactor can be held subcritical, but with the capability to return to full power within the period of time it takes to withdraw control rods. During this hot shutdown period, temperature is maintained at no-load T_{avg} by initially dumping steam to remove core residual heat, or at later stages, by running reactor coolant pumps to maintain system temperature.

Following shutdown, xenon buildup occurs and increases the degree of shutdown; i.e., initially, with initial xenon concentrations and all control rods inserted, the core is maintained at a minimum of 1 percent $\Delta k/k$ subcritical. The effect of xenon buildup is to increase this value to a maximum of about 4 percent $\Delta k/k$ at about 8 hours following shutdown from equilibrium full power conditions. If hot shutdown is maintained past this point, xenon decay results in a decrease in degree of shutdown. Since the value of the initial xenon concentration is about 3 percent $\Delta k/k$ (assuming that an equilibrium concentration had been reached during operation), boration of the reactor coolant is necessary to counteract the xenon decay and maintain shutdown.

If a rapid recovery is required, dilution of the system may be performed to counteract this xenon buildup. However, after the xenon concentration reaches a peak, boration must be performed to maintain the reactor subcritical as the xenon decays out.

4. Cold Shutdown

Cold shutdown is the operation which takes the reactor from hot shutdown conditions to cold shutdown conditions (reactor is subcritical by at least 1 percent $\Delta k/k$ and $T_{avg} \leq 200°F$).

Before initiating a cold shutdown, the RCS hydrogen concentration is lowered by reducing the volume control tank overpressure, by replacing the volume control tank hydrogen atmosphere with nitrogen, and by continuous purging to the Gaseous Waste Processing System.

Before cooldown and depressurization of the reactor plant is initiated, the reactor coolant boron concentration is increased to the cold shutdown value. After the boration is completed and reactor coolant samples verify that the concentration is correct, the operator resets the Reactor Makeup Control System for leakage makeup and system contraction at the shutdown reactor coolant boron concentration.

Contraction of the coolant during cooldown of the RCS results in actuation of the pressurizer level control to maintain normal pressurizer water level. The charging flow is increased, relative to letdown flow, and results in a decreasing volume control tank level.

The volume control tank level controller automatically initiates makeup to maintain the inventory.

After the RHRS is placed in service and the reactor coolant pumps are shutdown, further cooling of the pressurizer liquid is accomplished by charging through the auxiliary spray line. Coincident with plant cooldown, a portion of the reactor coolant flow is diverted from the RHRS to the CVCS for cleanup. Demineralization of ionic radioactive impurities and stripping of fission gases reduce the reactor coolant activity level sufficiently to permit personnel access for refueling or maintenance operations.

9.3.4.1.3 Safety Evaluation

The classification of structures, components and systems is presented in Section 3.2. A further discussion on seismic design categories is given in Section 3.7N. Conformance with NRC General Design Criteria for the plant systems, components and structures are important to safety as presented in Section 3.1. Also Appendix 1A(N) provides a discussion on applicable Regulatory Guides.

9.3.4.1.3.1 Reactivity Control

Any time that the plant is at power, the quantity of boric acid retained and ready for injection always exceeds that quantity required for the normal cold shutdown assuming that the control assembly of greatest worth is in its fully withdrawn position. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. An adequate quantity of boric acid is also available in the refueling water storage tank to achieve cold shutdown.

When the reactor is subcritical (i.e., during cold shutdown, hot shutdown, hot standby) the source range nuclear instrumentation continuously monitors the neutron flux to detect any indication of an inadvertent boron dilution transient. Upon the detection of a doubling of the neutron flux during any of the aforementioned modes of operation, an alarm is sounded to alert the operator. Other alarms in the CVCS system are also available to alert the operator of a potential inadvertent boron dilution transient. (See also Section 15.4.6 for a discussion of the inadvertent boron dilution transient).

The rate of boration, with a single boric acid transfer pump operating, is sufficient to take the reactor from full power operation to 1 percent shutdown in the hot condition, with no rods inserted, in less than 90 minutes. In less than 90 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 25 hours after shutdown. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions.

Two separate and independent flow paths are available for reactor coolant boration; i.e., the charging line and the reactor coolant pump seal injection line. A single failure does not result in the inability to borate the RCS.

If the normal charging line is not available, charging to the RCS is continued via reactor coolant pump seal injection at the rate of approximately 5 gpm per pump. At the charging rate of 20 gpm (5 gpm per reactor coolant pump), approximately 6.0 hours are required to add enough boric acid

solution to counteract xenon decay, although xenon decay below the full power equilibrium operating level will not begin until approximately 25 hours after the reactor is shutdown.

A third boration path can be provided, if the normal charging line or reactor coolant pump seal injection paths are not available. The reactor operator can cool down and depressurize the reactor coolant system to a pressure less than the shutoff head of the safety injection pumps without allowing the core to return to a critical condition. Borated water from the Refueling Water Storage Tank could then be delivered to the reactor coolant system via the safety injection pumps.

As backup to the normal boric acid supply, the operator can align the refueling water storage tank outlet to the suction of the charging pumps.

Since inoperability of a single component does not impair ability to meet boron injection requirements, plant operating procedures allow components to be temporarily out of service for repairs. However, with an inoperable component, the ability to tolerate additional component failure is limited. Therefore, operating procedures require immediate action to affect repairs of an inoperable component, restrict permissible repair time, and require demonstration of the operability of the redundant component.

9.3.4.1.3.2 Reactor Coolant Purification

The CVCS is capable of reducing the concentration of ionic isotopes in the purification stream as required in the design basis. This is accomplished during normal operation by passing the letdown flow through one of the mixed bed demineralizers which removes ionic isotopes, except those of cesium, molybdenum and yttrium, with a minimum decontamination factor of 10. Through occasional use of the cation bed demineralizer the concentration of cesium can be maintained below 1.0 c/cc, assuming 1 percent of the rated core thermal power is being produced by fuel with defective cladding. The cation bed demineralizer is capable of passing a maximum purification letdown flow of 170 gpm, though only a portion of this capacity is normally utilized. Each mixed bed demineralizer is also capable of processing more than the maximum normal purification letdown flow rate. If the operating mixed bed demineralizer's resin has become exhausted, the second demineralizer can be placed in service. During RCS cleanup for outages, the mixed bed demineralizers may be placed in parallel service to allow increased flow and provide a more rapid cleanup. Each demineralizer is capable of being operated for one core cycle with 1 percent defective fuel.

A bypass line and isolation valve also provide additional flexibility for cleanup letdown flow. The line and valve allow letdown flow to be aligned such that the mixed bed demineralizers are bypassed and flow is directly to the cation bed demineralizer for cleanup. Letdown can also be aligned to flow to the BTRS demineralizer(s) for cleanup, either directly or in series with either a mixed bed or the cation bed demineralizer.

A further cleanup feature is provided for use during Residual Heat Removal operations. A RHR letdown flow control valve admits flow into the CVCS downstream of the containment isolation valves.

The maximum temperature that will be allowed for the letdown stream demineralizers is approximately 140°F. If the temperature of the letdown stream approaches this level, the flow will be automatically diverted so as to bypass the demineralizers. If the letdown is not diverted, the

only consequence would be a decrease in ion removal capability. Ion removal capability starts to decrease when the temperature of the resin goes above approximately 160°F for anion resin or above approximately 250°F for cation resin. The resins do not lose their exchange capability immediately. Ion exchange still takes place (at a faster rate) when temperature is increased. However, with increasing temperature, the resin loses some of its ion exchange sites along with the ions that are held at the lost sites. The ions lost from the sites may be reexchanged farther down the bed. The number of sites lost is a function of the temperature reached in the bed and of the time the bed remains at the high temperature. Capability for ion exchange will not be lost until a significant portion of the exchange sites are lost from the resin.

There would be no safety problem associated with overheating of the demineralizer resins. The only effect on reactor operating conditions would be the possibility of an increase in the reactor coolant activity level. If the activity level in the reactor coolant were to exceed the limit given in the Technical Specifications, reactor operation would be restricted as required by the Technical Specifications.

9.3.4.1.3.3 Seal Water Injection

Flow to the reactor coolant pump seals is assured by the fact that there are three charging pumps, anyone of which is capable of supplying the normal charging line flow plus the nominal seal water flow.

9.3.4.1.3.4 Hydrostatic Testing of the Reactor Coolant System

The positive displacement pump can pressurize the RCS to its maximum specified hydrostatic test pressure. The pump is capable of producing a hydrostatic test pressure greater than that required.

9.3.4.1.3.5 Leakage Provisions

CVCS components, valves, and piping which see radioactive service are designed to limit leakage to the atmosphere. The following are preventive means which are provided to limit radioactive leakage to the environment.

1. Where pressure and temperature conditions permit, diaphragm type valves are used to essentially eliminate leakage to the atmosphere.

2. All packed valves which are larger than 2 inches and which are designated for radioactive service are provided with a stuffing box and lattern leakoff connections.

3. All control (modulating) and three-way valves are either provided with stuffing box and leakoff connections or are totally enclosed.

4. Welding of all piping joints and connections except where flanged connections are provided to facilitate maintenance and hydrostatic testing.

The volume control tank provides an inferential measurement of leakage from the CVCS as well as the RCS. The amount of leakage can be inferred from the amount of makeup added by the Reactor Makeup Control System.

During normal operation, the hydrogen and fission gases in the volume control tank are purged to the Waste Processing System, to limit the release of radioactive gases through leakage by maintaining the radioactive gas level in the reactor coolant several times lower than the equilibrium level. Also, provided are demineralizers which maintain reactor coolant purity, thus reducing the radioactivity level of the RCS water.

9.3.4.1.3.6 Ability to Meet the Safeguards Function

A failure analysis of the portion of the CVCS which is safety-related (used as part of the Emergency Core Cooling System) is included as part of the Emergency Core Cooling System failure analysis presented in Tables 6.3-5 and 6.3-6.

9.3.4.1.3.7 Heat Tracing

Heat tracing requirements for boric acid solutions depends mainly on the solution concentration. For this plant the concentration of boric acid ranges from 10 ppm to 4 weight percent boric acid. Electrical heat tracing is not required on any CVCS components which contain 4 weight percent boric acid, providing these components are located in a room maintained at 65°F or higher. Redundant temperature alarms are provided to assure room temperature does not go below 65°F. Refer to Section 9.3.4.1.2 for more information.

9.3.4.1.3.8 Abnormal Operation

The CVCS is capable of making up for an RCS leak of approximately 130 gpm using one centrifugal charging pump and still maintaining seal injection flow to the reactor coolant pumps. This also allows for a minimum RCS cooldown contraction. This is accomplished with the letdown isolated.

9.3.4.1.3.9 Failure Mode and Effects Analysis

A failure mode and effects analysis (FMEA) of the Chemical and Volume Control System is provided as Table 9.3-9.

9.3.4.1.4 Tests and Inspections

As part of plant operation, periodic tests, surveillance inspections and instrument calibrations are made to monitor equipment condition and performance. Most components are in use regularly; therefore, assurance of the availability and performance of the systems and equipment is provided by Control Room and/or local indication.

Refer to Chapter 14 for further information.

9.3.4.1.5 Instrumentation Application

Process control instrumentation is provided to acquire data concerning key parameters about the CVCS. The location of the instrumentation is shown on Figure 9.3-10.

The instrumentation furnishes input signals for monitoring and/or alarming purposes. Indications and/or alarms are provided for the following parameters:

1. Temperature.

2. Pressure.

3. Flow.

4. Water level.

The instrumentation also supplies input signals for control purposes. Some specific control functions are:

1. Letdown flow is diverted to the volume control tank upon high temperature indication upstream of the mixed bed demineralizers.

2. Pressure upstream of the letdown heat exchanger is controlled to prevent flashing of the letdown liquid.

3. Charging flow rate is controlled during charging pump operation.

4. Water level is controlled in the volume control tank.

5. Temperature of the boric acid solution in the batching tank is maintained.

6. Reactor makeup is controlled.

7. Temperature of letdown flow to the Boron Thermal Regeneration System is controlled.

8. Temperature of the chilled water flow to the letdown chiller heat exchanger is controlled.

9. Temperature of letdown flow return from the boron thermal regeneration demineralizers is controlled.

9.3.4.2 Boron Recycle System

The Boron Recycle System (BRS) receives and recycles reactor coolant effluent for possible reuse of the boric acid and makeup water. The system decontaminates the effluent by means of demineralization and gas stripping, and uses evaporation to separate and recover the boric acid and makeup water.

9.3.4.2.1 Design Bases

9.3.4.2.1.1 Collection Requirements

The BRS collects and processes effluent which may be reused as makeup to the RCS. For the most part, this effluent is the deaerated, tritiated, borated, and radioactive water from the letdown and process drains.

The BRS was designed to collect, via the letdown line in the CVCS, the excess reactor coolant that results from the following plant operations during one annual core cycle. Because it is

operated in batches, the effluent from longer core cycles can be processed. The inputs used to design the system are:

1. Dilution for core burnup from approximately 1200 ppm boron at the beginning of a core cycle to approximately 10 ppm near the end of the core cycle.

2. Hot shutdowns and startups. Four hot shutdowns are assumed to take place during an annual core cycle.

3. Cold shutdowns and startups. Three cold shutdowns are assumed to take place during an annual core cycle.

4. Refueling shutdown and startup.

The BRS may also collect water from the following sources:

1. Reactor coolant drain tank (Liquid Waste Processing System) - collects leakoff type drains from equipment inside the Containment.

2. Volume control tank and charging pump suction pressure reliefs (CVCS) and safety injection pumps pressure reliefs (Emergency Core Cooling System).

3. Boric acid blender (CVCS) - provides storage of boric acid if a boric acid tank must be emptied for maintenance. The boric acid solution is stored in a recycle holdup tank after first being diluted with reactor makeup water by the blender to prevent precipitation of the boric acid in the unheated recycle holdup tank.

4. Accumulators (Safety Injection System) - collects effluent resulting from leak testing of accumulator check valves.

5. Liquid Waste Processing System - provides capability for using the recycle evaporator as a waste evaporator and vice versa.

6. Spent fuel pool pumps (Spent Fuel Pool Cooling and Cleanup System) - provides a means of storing the fuel transfer canal water in case maintenance is required on the transfer equipment.

7. Valve leakoffs and equipment drains.

9.3.4.2.1.2 Capacity Requirement

The BRS is designed to process the total volume of water collected during a core cycle as well as short term surges. The design surge is that produced by a cold shutdown and subsequent startup during the latter part of a core cycle or by a refueling shutdown and startup.

9.3.4.2.1.3 Purification Requirement

The water collected by the BRS contains dissolved gases, boric acid, and suspended solids. Based on reactor operations with 1 percent of the rated core thermal power being generated by fuel elements with defective cladding, the BRS is designed to provide sufficient cleanup of the

water to satisfy the chemistry requirements of the recycled reactor makeup water and 4 weight percent boric acid solution.

The maximum radioactivity concentration buildup in the BRS components is based on operation of the reactor at its engineered safeguards design rating with defective fuel rods generating 1 percent of the core thermal power. For each component, the shielding design considers the maximum buildup on an isotopic basis including only those isotopes which are present in significant amounts. Filtration, demineralization, and evaporation are the means by which the activity concentrations may be controlled.

9.3.4.2.2 System Description

The BRS is shown in Figure 9.3-11 (piping and instrumentation diagram). The codes and standards to which the individual components of the BRS are designed are listed in Section 3.2. When water is directed to the BRS, the flow passes first through the recycle evaporator feed demineralizers and filters and then into the recycle holdup tanks. The recycle evaporator feed pumps can be used to transfer liquid from one recycle holdup tank to the other if desired. When sufficient water is accumulated to warrant evaporator operation, the recycle evaporator feed pumps take suction from the selected recycle holdup tank. The fluid then flows through the recycle evaporator. Here, dissolved gases (i.e., hydrogen, fission gases and other gases) are removed in the stripping column before the liquid enters the evaporator shell. These gases are directed to the evaporator vent condenser and then to the plant ventilation system. An alternate flow path to the Gaseous Waste Processing System is available.

During evaporator operation, distillate from the evaporator flows to the Waste Monitor Tanks for discharge. Also located in this flow path, are the recycle evaporator condensate demineralizer and the recycle evaporator condensate filter.

The evaporator concentrates the boric acid solution until approximately 4 weight percent solution is obtained. The accumulated batch is normally transferred directly to the boric acid tanks in the CVCS through the recycle evaporator concentrates filter. Before transferring the boric acid from the evaporator to the boric acid tanks, it is analyzed, and, if it does not meet the required chemical standards, it can be diverted back to the recycle holdup tanks for reprocessing or to the Liquid Waste Processing System for disposal.

Connections are provided so that, if necessary, the recycle evaporator can be used as a waste evaporator (and vice versa).

All portions of the BRS which contain concentrated boric acid solution are located within a heated area in order to maintain solution temperature at ≥65°F. This is 10°F above the solubility limit for the nominal 4 weight percent boric acid solution. If a portion of the system which normally contains concentrated boric acid solution is not located in a heated area, it must be provided with some other means (e.g., heat tracing) to maintain solution temperature ≥65°F.

The BRS can also provide a direct source of RHUT water to fill the RWSTs. A BR cross tie downstream of the Recycle Evaporator Feed Pumps provides a flowpath through SF piping, Safety Injection valves 1 & 2-8800A & B, and on to the RWSTs through the ECCS suction headers. This flowpath provides a quicker transfer of water than the flowpath through process equipment and the Spent Fuel Pools/Transfer Canal. Before transferring the RHUT water, it is sampled and analyzed to ensure it meets specified chemical standards. This will prevent the

RHUT water from diluting the RWSTs' boron concentration or contaminate the RWSTs. See Figures 9.1-13 and 9.3-11.

9.3.4.2.2.1 Component Descriptions

A summary of principal component data is given in Table 9.3-8 and the code requirements are given in Section 3.2.

Recycle Evaporator Feed Pumps

Two centrifugal, canned pumps supply feed to the recycle evaporator package from the recycle holdup tanks. The pumps can also be used to recirculate water from the recycle holdup tanks through the recycle evaporator feed demineralizers for cleanup if desired. An auxiliary discharge connection is provided to return water to the transfer canal from the recycle holdup tanks, if those tanks were used for storage of transfer canal water during refueling equipment maintenance. An auxiliary discharge connection is provided to supply water to the suction of the charging pumps (CVCS) for refilling the RCS after loop or system drain. An auxiliary discharge connection is provided to allow processing of the recycle holdup tank water to the Filter Demineralizer System.

The Recycle Evap Feed Pumps can also be used to transfer RHUT water directly to the RWSTs.

Recycle Holdup Tanks

Two recycle holdup tanks provide storage of radioactive fluid which is discharged from the RCS during startup, shutdown, load changes and boron dilution. The tanks are constructed of austenitic stainless steel.

Each tank has a diaphragm which prevents air from dissolving in the water and prevents the hydrogen and fission gases in the water from mixing with the air. The volume in the tank above the diaphragm is continuously ventilated with building supply air, and the volume of gas below the diaphragm is intermittently vented to the Waste Processing System (Gas).

In addition to the collection of effluent as described above, the recycle holdup tanks provide the following functions:

1. Serve as a head tank for the recycle evaporator feed pumps.

2. Provide holdup for a RCS drain to the centerline of the reactor vessel nozzles, including the pressurizer and steam generators.

3. Provide storage for transfer canal water during refueling equipment maintenance.

4. Collect discharge from relief valves in borated water systems outside of containment.

5. Provide storage of water for the RWSTs.

Recycle Evaporator Reagent Tank

This tank provides a means of adding chemicals to the evaporator for process treatment.

Recycle Evaporator Feed Demineralizers

Two flushable, mixed bed demineralizers remove fission products from the fluid directed to the recycle holdup tanks. The demineralizers also provide a means of cleaning the recycle holdup tank contents via recirculation.

Recycle Evaporator Condensate Demineralizer

A flushable demineralizer is provided as a polishing demineralizer for distillate from the recycle evaporator. Although the bed may become saturated with boron at the normally low concentration (<10 ppm) leaving the evaporator, it will still remove boron if the concentration increases because of an evaporator upset. The demineralizer also provides a means of cleanup of the Rector Makeup Water Storage Tank contents.

Recycle Evaporator Feed Filter

This filter collects resin fines and particulates from the fluid entering the recycle holdup tanks.

Recycle Evaporator Condensate Filter

This filter collects resin fines and particulate from the boric acid evaporator condensate stream.

Recycle Evaporator Concentrates Filter

This filter removes particulates from the evaporator concentrate as it leaves the evaporator.

Recycle Evaporator

The recycle evaporator package processes dilute boric acid and produces distillate and approximately 4 weight percent boric acid stripped of hydrogen, radioactive gases, and other dissolved gases.

A boric acid solution is fed from the recycle holdup tanks to the evaporator by the recycle evaporator feed pumps. The feed first passes through a heat exchanger where auxiliary steam raises its temperature. The feed then passes into the top of the stripping column. Gases are stripped off as the feed passes over the packing in the tower in contra flow to stripping steam from the evaporator. After stripping, the feed is introduced into the evaporator as makeup. The vapors leaving the boiling pool are stripped of entrained liquid and volatile boron carryover. Pure vapors are then condensed in the condenser section and pumped from the system. When the desired concentration is reached in the boiling pool, the concentrates are pumped from the system.

Radioactive gases, hydrogen, and other non-condensables are discharged from the system into the plant ventilation system.

The recycle and waste evaporators are interconnected so that they can serve as standbys for each other under abnormal conditions.

Recycle Holdup Tank Vent Ejector

The ejector is designed to pull gases from under the diaphragm in a recycle holdup tank and deliver them to the Gaseous Waste Processing System. Nitrogen, provided by the standby waste gas compressor, provides the motive force.

9.3.4.2.2.2 System Operation

The BRS is manually operated with the exception of a few automatic protection functions. These automatic functions protect the recycle evaporator feed demineralizers from a high inlet temperature and a high differential pressure, prevent a high vacuum from being drawn on the recycle holdup tank diaphragm, protect the recycle evaporator feed pumps from low net positive suction head, and prevent high activity recycle evaporator condensate from being sent to the Reactor Makeup Water Storage Tank. The BRS has sufficient instrumentation readouts and alarms to provide the operator information to assure proper system operation.

Evaporation

Water is accumulated in the recycle holdup tank until sufficient quantity exists to warrant an evaporator startup. Prior to startup of the evaporator, the contents of the recycle holdup tank are analyzed and, if necessary, are recirculated through the recycle evaporator feed demineralizers and filter. The flow can be discharged back to the recycle holdup tank or to the evaporator. The evaporator is then operated to produce a batch of 4 weight percent boric acid.

During the operation of the evaporator, condensate may be routed to a Waste Monitor Tank or may be continuously sent to the Recycle Holdup Tank via the recycle evaporator condensate demineralizer.

After a batch of boric acid is concentrated to 4 weight percent, it is analyzed to ensure that it is within specifications for reuse. If it meets the specifications, it is pumped to the boric acid tanks. If it does not, it can be returned to the recycle holdup tank via the recycle evaporator feed demineralizers for reevaporation or, if desired, the concentrated boric acid can be sent to the Liquid Waste Processing System for disposal.

Storage of Water for the RWSTs

The BRS can be used to store water for the RWSTs. A BR cross tie can transfer RHUT water through a small portion of the SF system, through the 1 & 2-8800A & B valves, and on to either RWST. The Recycle Evap Feed Pumps provide the pumping force for this transfer which allows the bypassing of BRS process equipment and the SF pools/Transfer Canal. This results in a more efficient transfer of the water. The water in the RHUTs must be sampled to meet appropriate chemistry requirements prior to the transfer. See Figures 9.1-13 and 9.3-11.

Recycle Holdup Tank Venting

Because hydrogen is dissolved in the reactor coolant at approximately 1 atmosphere overpressure, a portion of the hydrogen along with fission gases will come out of solution in the recycle holdup tank under the diaphragm. The hydrogen and fission gases are vented to the Gaseous Waste Processing System as required. The total integrated flow from the letdown line and the reactor coolant drain tank to the recycle holdup tanks is monitored. An alarm indicates

when a sufficient amount of water has passed to the recycle holdup tanks to require venting of the accumulated gases. A recycle holdup tank will be vented after a loop drain or a fuel storage area drain as per the plant operating instructions.

When venting of either recycle holdup tank is required, the following steps are observed:

1. All inlets to the recycle holdup tank are closed.

2. The recycle holdup tank is emptied of water to a low level by either processing with the evaporator or transferring to the other recycle holdup tank.

3. The standby waste gas compressor is lined up to the recycle holdup tank vent ejector. Normally, the standby compressor will feed the other waste gas compressor which is lined up to a catalytic recombiner and a high activity gas decay tank. However, in the event of a recycle holdup tank diaphragm leak or after a RCS loop drain or spent fuel pool drain, a shutdown gas decay tank is used instead of a high activity gas decay tank. This prevents accumulation of air (i.e., nitrogen) in the high activity gas decay tanks.

4. The standby gas compressor is started up and the vent from the holdup tank is opened. The vent flow is throttled to approximately 1 scfm. At this time, a sample of the vent gases can be taken to check the composition.

5. When the gases have been vented from the recycle holdup tank, the pressure in the vent line decreases, which automatically trips the recycle holdup tank vent isolation valve closed.

6. After the vent isolation valve closes, the manual vent valve is closed, the gas compressor is shut down, and the recycle holdup tank inlets and outlets are lined up for normal use.

Maintenance Drains

When large amounts of water must be drained from the RCS or the fuel storage area (or fuel transfer canal) to the BRS, a recycle holdup tank is drained of water and vented to the Gaseous Waste Processing System.

The water can then be stored in this tank until maintenance is completed and, after checking the chemistry, returned. After returning the water, the recycle holdup tank is again vented to the Gaseous Waste Processing System.

Reactor Makeup Water Cleanup

If the reactor makeup water requires purification, it can be recirculated through the recycle evaporator condensate demineralizer until its chemistry is within specifications. If further processing is necessary, water from the Reactor Makeup Water Storage Tank can be directed through the recycle evaporator condensate demineralizer and into the recycle holdup tanks for re-evaporation.

Waste Processing with the Recycle Evaporator

Connections are provided so that the recycle evaporator can be used as a waste evaporator (and vice versa). These consist of a feed connection from the waste holdup tank, a condensate connection to the waste condensate tank, a concentrate connection to the Plant Effluent Tank Room, and an evaporator vent connection to the plant vent. Thus, the recycle evaporator can be used to perform the function of the waste evaporator except that, since heat tracing is not provided for the recycle evaporator, the boric acid would be concentrated to no more than 4 weight percent rather than to 12 weight percent, as is permitted for the waste evaporator.

After using the recycle evaporator to process water from the Liquid Waste Processing System, it is thoroughly rinsed out. During initial recycle processing, the condensate is directed to the waste condensate tank for analysis prior to recycling to the reactor makeup water storage tank. Depending upon the purity of the evaporator bottoms, the concentrated boric acid can be recycled to the boric acid tanks, returned to the recycle holdup tanks for reprocessing, or transferred to the Liquid Waste Processing System for disposal.

9.3.4.2.3 Safety Evaluation

Malfunctions in the BRS do not affect the safety of station operations. The BRS is designed to tolerate equipment faults with criterial functions being met by the use of the two pieces of equipment so that the failure of one will, at most, reduce the capacity of the BRS but not completely shut it down. Because of the large surge capacity of the BRS, the non-availability of the recycle evaporator can be tolerated for periods of time. Also, backup is provided by the waste evaporator.

9.3.4.2.4 Tests and Inspections

The BRS is in intermittent use throughout normal reactor operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice. Refer to Chapter 14 for further information.

9.3.4.2.5 Instrumentation Application

The instrumentation available for the BRS is discussed below. Alarms are provided as noted. There is also a common alarm on the main control board which indicates any alarms on the BRS panel.

9.3.4.2.5.1 Temperature

Temperature at Inlet to the Recycle Evaporator Feed Demineralizer

Instrumentation is provided to measure the temperature of the inlet flow to the recycle evaporator feed demineralizers and to control a three-way bypass valve. If the inlet temperature becomes too high, the instrumentation aligns the valve to bypass the demineralizers. Local temperature indication and a high temperature alarm on the BRS panel are provided by this instrumentation.

9.3.4.2.5.2 Pressure

Pressure Differential Across the Recycle Evaporator Feed Demineralizers

Instrumentation is provided to measure the pressure differential across the recycle evaporator feed demineralizers and to control the same three-way valve as discussed above (but independently of the temperature control). If the pressure drop through the demineralizers is too high, this instrumentation aligns the valve to divert flow directly to the recycle evaporator feed filters. Local pressure differential indication and a high alarm on the BRS panel are provided by this instrumentation.

Pressure at Inlet and Outlet of Filters

Instrumentation is provided to measure the pressure differential across each recycle evaporator feed filter, the recycle evaporator concentrates filter, and the recycle evaporator condensate filter. Local indication of the pressure in each inlet and outlet line is provided.

Pressure at Discharge of Recycle Evaporator Feed Pump

Instrumentation is provided to measure and give local indication of the discharge pressure of each recycle evaporator feed pump.

Pressure in Vent Line from the Recycle Holdup Tanks

Instrumentation is provided to measure the pressure in the recycle holdup tank vent line and to control a shutoff valve in the vent line. This instrumentation is used during holdup tank venting operations. When the pressure in this line becomes too low, the valve will be automatically closed to protect the holdup tank diaphragm from an excessive differential pressure across it. Local pressure indication and low pressure alarm on the BRS panel are provided.

9.3.4.2.5.3 Flow

Boron Recycle System Flow Totalizer

Instrumentation is provided to monitor the total integrated flow received by the BRS from the letdown line (CVCS) and the reactor coolant drain tank (Waste Processing System). Indication of integrated flow and high alarm are given on the BRS panel. Actuation of the high alarm indicates that the integrated flow has reached a value at which the volume of gases (hydrogen and fission gases) which have come out of solution should be vented from the recycle holdup tanks.

Flow in Vent Line from Recycle Holdup Tanks

Instrumentation is provided which gives local indication of the recycle holdup tank vent purge flow.

Flow in Feed Line to the Recycle Evaporator

Instrumentation is provided which gives local indication of recycle evaporator feed flow.

9.3.4.2.5.4 Level

Level in the Recycle Holdup Tanks

Instrumentation is provided to give an indication of the water level of each recycle holdup tank. Both high level and low level alarms are provided by this instrumentation at the BRS panel. The recycle evaporator feed pumps are stopped on the holdup tank low level signal.

9.3.4.2.5.5 Deleted

REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, Design Bases for Protection Against Natural Phenomena.

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, Environmental and Missile Design Bases.

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, Sharing of Structures, System, and Components.

4. 10 CFR Part 50, Appendix A, General Design Criterion 56, Primary Containment Isolation.

5. NRC Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste- Containing Components of Nuclear Power Plants, U.S. Nuclear Regulatory Commission.

6. NRC Regulatory Guide 1.29, Seismic Design Classification, Revision 1, U.S. Nuclear Regulatory Commission.

7. Branch Technical Position APCSB 3-1, Protection Against Postulated Failures in Fluid Systems Outside Containment.

8. Branch Technical Position MEB 3-1, Postulated Break and Leakage Locations in Fluid System Piping Outside Containment.

9. ANSI N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, 1973.

10. NRC Regulatory Guide 8.8, Information Relevant to Maintaining Occupational Radiation Exposure As Low As Reasonably Achievable (Nuclear Power), Revision 1, Sept. 1975, U. S. Nuclear Regulatory Commission

TABLE 9.3-1
INSTRUMENT AIR SYSTEM COMPONENT CHARACTERISTICS

(Sheet 1 of 2)

A. DELETED

B. DELETED

C. ROTARY AIR COMPRESSOR PACKAGES AND ASSOCIATED EQUIPMENT

1. Compressor Packages

Number of Units Required	Four rotary screw and two rotary lobe, two stage, oil-free, water cooled Compressors
Safety Class	NNS
Operating Mode	Lead/Backup
Capacity at Design Inlet Conditions (14.5 psia, 122°F, 100% Relative Humidity)	689 ACFM
Capacity at Standard Inlet Conditions (14.7 psia, 60°F, 36% Relative Humidity).	650 SCFM
Motor, hp	200
Compressor Speed, rpm	3580
Intercooler, Aftercooler, & Oil Cooler Exit Cooling Water Temperature based on 95°F Inlet, °F	125 122 (Common and Unit 2)
Total Cooling Water Requirement, gpm	30 (Unit 1) 40 (Common and Unit 2)

2. Instrument Air Receivers

Number of Receivers Required	3 Vertical Cylindrical Tank and 3 Horizontal Cylindrical Tank
Operating Mode	1 Receiver Dedicated to each Air Compressor Package
Design Pressure, psig	125 165 (Common X-01)
Design Temperature, °F	150 (Vertical) 250 (Horizontal) 450 (Horizontal X-01)

TABLE 9.3-1
INSTRUMENT AIR SYSTEM COMPONENT CHARACTERISTICS

(Sheet 2 of 2)

	Volume, ft^3	200 (Vertical) 222 (Horizontal) 204 (Horizontal X-01)
3.	Instrument Air Dryer Packages	
	Number of Dryers Required	6 Dual Tower, Internal Heater, Regenerative Type
	Safety Class	NNS
	Operating Mode	1 Dryer Package Dedicated to each Air Compressor Package
	Capacity, SCFM	1000

Notes:

Aftercoolers are integral to each air compressor package.

Pre & Afterfilters are integral to each dryer package.

Pre & Afterfilters are integral to each dryer.

TABLE 9.3-2
SERVICE AIR SYSTEM COMPONENT CHARACTERISTICS

1. Service Air Compressor Package

Number of units required	1, Single stage, oil injected air cooled, rotary screw compressor
Safety class	NNS
Operating mode	1, 100-percent-capacity unit
Capacity, ft^3/min (29.07 in.Hg abs, 104^oF, and 70 percent relative humidity)	1300, full load
Motor size, hp	250
Compressor, bhp	270
Volts/Hz/phase	460/60/3

2. Service Air Receiver

Number of units required cylindrical	1, horizontal, receiver
Safety class	NNS
Operating mode	1, 100-percent-capacity unit
Maximum operating pressure, psig	107
Ambient temperature, ^oF	125

TABLE 9.3-3
FAILURE MODE AND EFFECT ANALYSIS FOR AIR-OPERATED SAFETY-RELATED VALVES

(Sheet 1 of 4)

Valve Designation	System	Failure Mode Note 1	Effect On Plant Safety	Remarks
Power-operated relief Valve	MS	Closed	No effect	Accumulators are provided for SGTR mitigation or to permit remote manual control for up to 4 hours to cooldown from hot standby at 50°F hour, followed by manual operator control if required to cooldown RHRS conditions.
Main Steam Isolation	MS	None	No effect	The MSIVs fail closed on loss of valves hydraulic fluid (See Section 10.3.2.3). The hydraulic pumps are air driven.
Hydrazine and ammonia hydroxide feed isolation valves	CFS	Closed	No effect	
Motor-driven AF pumps recirculation valve	AF	Open	No effect	Accumulators provided to permit automatic control for a 30 minute period. Manual values may be used to throttle or isolate recirculation flow locally as required.
Motor-driven AF pumps discharge valves	AF	Open	No effect	Required AF flow ensured to the steam generators. Manual operators provided at valve locations. Accumulators provided to permit remote manual isolation control for a 30-min period.
Turbine-driven AF pumps discharge valves	AF	Open	No effect	Required AF flow ensured to the steam generators. Manual operators provided at valve locations. Accumulators provided to permit remote manual isolation for a 30-min period.
Component cooling water pumps recirculation valves	CCWS	Closed	No effect	Maximum flow to equipment requiring cooling

TABLE 9.3-3
FAILURE MODE AND EFFECT ANALYSIS FOR AIR-OPERATED SAFETY-RELATED VALVES

(Sheet 2 of 4)

Valve Designation	System	Failure Mode Note 1	Effect On Plant Safety	Remarks
Demineralized water supply to CCWS surge tank valve	CCWS	Open	No effect	
Reactor makeup water to CCWS surge tank valve	CCWS	Open	No effect	Maximum makeup water supply ensured to the surge tank. Valves have manual operators.
Safety Chiller Condenser CCWS Regulating Valves	CCWS	Open	No effect	Valves are backed up by Accumulator air supply for a minimum of 30 minutes. They are also equipped with handwheels for manual action.
Pressurizer spray valves	RCS	Closed	No effect	
RHR heat exchanger discharge valves	RHR	Open	No effect	Maximum flow ensured through the RHR heat exchangers.
Containment isolation	Various	Closed	No effect	Valves
Charging pump suction high point vent	CS	Closed	No effect	Fails closed on loss of power, generation of a VCT Lo-Lo water level, generation of a source range neutron flux doubling signal and the generation of a safety injection “S” signal.
CCWS surge tank level control valve	CCWS	Closed	No effect	Prevent flooding of CCWS surge tank. Handwheels provided for manual control.
Process Sampling System Control and Isolation valves	PSS	Closed	No effect	

TABLE 9.3-3
FAILURE MODE AND EFFECT ANALYSIS FOR AIR-OPERATED SAFETY-RELATED VALVES

(Sheet 3 of 4)

Valve Designation	System	Failure Mode Note 1	Effect On Plant Safety	Remarks
MS to AF pump turbine isolation valve	MS	Open	No effect	Maximum steam flow to AF pump turbine. Accumulators provided to permit remote manual isolation for a seven hour period.
Blowdown Control valves	MS	Open	No effect	Valves upstream of the isolation valves
Blowdown isolation valves	MS	Closed	No effect	Valves of the Containment Isolation System.
Steam Generator sample control valves	MS	Closed	No effect	Used to select upper or lower sample
Steam generator sample isolation valves	MS	Closed	No effect	Valves of the Containment Isolation System.
Main steam drain pot isolation valve	MS	Closed	No effect	Valves of the Containment Isolation System.
Demineralizer makeup to condensate storage tank isolation valve	AF	Closed	No effect	Makeup not required during an accident
Letdown HX outlet CCWS return control valve	CCWS	Open	No effect	Valves of the CCWS Non-safeguard Loop
Spent fuel pool HX CCWS return control valve	CCWS	Open	No effect	Maximum cooling to spent fuel pool HX

TABLE 9.3-3
FAILURE MODE AND EFFECT ANALYSIS FOR AIR-OPERATED SAFETY-RELATED VALVES

(Sheet 4 of 4)

Valve Designation	System	Failure Mode Note 1	Effect On Plant Safety	Remarks
Ventilation chillers CCWS supply and discharge control valves	CCWS	Closed	No effect	Valves of the CCWS Non-safeguard Loop
Excess letdown CCWS return isolation valves	CCWS	Closed	No effect	Valves of the CCWS Non-safeguard Loop
CCWS supply to excess Letdown and RCS drain tank	CCWS	Closed	No effect	Valves of the CCWS Non-safeguard Loop
Chemical additive tank	CSS	Open	No effect	Maximum amount of chemicals added during injection phase of control valve discharge safety injection
Reactor makeup water to safety chilled water surge tank isolation valves	CWS	Open	No effect	
Demineralized water to safety chilled water surge tank isolation valves	CWS	Closed	No effect	
Safety chilled water surge tank level control valves	CWS	Closed	No effect	Prevents flooding of surge tank. Has manual bypass

NOTES:

1: Failure Mode on loss of air supply

AFS - Auxiliary Feedwater System CCWS - Component Cooling Water System CFS - Chemical Feed System CSS - Containment Spray System CWS - Chilled Water System HX - Heat Exchanger MSSS - Main Steam Supply System PSS - Primary Sampling System SSWS - Station Service Water System

TABLE 9.3-4
PROCESS SAMPLING SYSTEM PARAMETERS

(Sheet 1 of 3)

Sample Point Description	Sampling Conditions: Design Temp. (F)	Sampling Conditions: Design Pressure (Psig)	Method of Sampling[a]: Grab Sample	Method of Sampling[a]: Sample Vessel	Method of Sampling[a]: Continuous Analyzers	Sample Flow (gpm): Maximum (Purge)	Sample Flow (gpm): Minimum (Purge/Grab Sample)	Parameter Measured[b]	Process Fluid	Sample Purpose	Application
SIS accumulator tanks	300[c]	700	X	-	-	1.0	0.264/0.15	B Cl	Reactor coolant	Detect deviation from specified boron and halide concentrations	Guide operation to prevent chloride stress corrosion
RCS pressurizer steam	650	2485	X	X	-	1.0	0.6/0.15	-	Saturated steam without boric gas	Detect accumulation of gross fission gas activity before shutdown	Schedule venting for controlled disposal
RCS pressurizer liquid space	650	2485	X	X	-	1.0	0.75/0.15	B SC pH C1	Reactor coolant	Detect deviation from specified values for conductivity, pH, total alkali, halide	Insure effective corrosion control prevent chloride, stress corrosion
RCS hot legs	650	2485	X	X	-	0.85	0.75/0.15	B SC pH C1	Reactor coolant	Detect deviations in reactor coolant chemistry and detect accumulation of fission product activity	Insure effective corrosion control maintain sufficient, hydrogen control, and guide system operation.
RHR (RHR pump bypass)	400	600	X	X	-	1.0	0.75/0.15	-	-	-	-
Upstream from thermal regeneration demineralizer	250[c]	300	X	-	-	1.0	-	B	-	Detect boron concentration	Monitor and evaluate demineralizer effectiveness

TABLE 9.3-4
PROCESS SAMPLING SYSTEM PARAMETERS

(Sheet 2 of 3)

Sample Point Description	Sampling Conditions		Method of Sampling[a]			Sample Flow (gpm)		Parameter Measured[b]	Process Fluid	Sample Purpose	Application
	Design Temp. (F)	Design Pressure (Psig)	Grab Sample	Sample Vessel	Continuous Analyzers	Maximum (Purge)	Minimum (Purge/Grab Sample)				
Downstream of thermal regeneration demineralizer	250[c]	300	X	-	-	1.0	-	B	-	Detect boron concentration	Monitor and evaluate demineralizer effectiveness
Volume control tank gas phase	250[c]	300	X	-	-	-	-	-	Hydrogen cover gas	Detect accumulation of gross fission production gas activity	Guide to venting and purging to control gaseous activity in reactor coolant
CVCS letdown line upstream from purification demineralizer	250[c]	300	X	-	-	1.0	-	Various	Reactor coolant	Reactor coolant chemistry	Backup to reactor coolant hot-leg
	250[c]	300	-	-	X	Varies	Varies	H_2/O_2	Reactor coolant	Reactor coolant chemistry	Backup to reactor coolant hot-leg
CVCS letdown line downstream of purification demineralizer	250[c]	300	X	-	-	1.0	-	C1	Reactor coolant	Provide reactor coolant activity reduction	Monitor decontamination factor
Steam generator blowdown (steam generator sample taps)	600	1300	X	-	X	1.0	0.4/0.15	CC SC Na R pH	Secondary system (steam generator)	Detect deviation from specified chemical technical specification	Guide to regulation of blowdown rate, and steam generator primary-to-secondary side leakage

TABLE 9.3-4
PROCESS SAMPLING SYSTEM PARAMETERS

(Sheet 3 of 3)

Sample Point Description	Sampling Conditions		Method of Sampling[a]			Sample Flow (gpm)		Parameter Measured[b]	Process Fluid	Sample Purpose	Application
	Design Temp. (F)	Design Pressure (Psig)	Grab Sample	Sample Vessel	Continuous Analyzers	Maximum (Purge)	Minimum (Purge/Grab Sample)				
Spent fuel pool demineralizers (only Unit 1)	150	150	X	-	-	1.0	0.75/0.15	B	Spent fuel pool fluid	Detect deviation from water chemistry specification	Determine resin exhaustion and performance
Downstream blowdown cation demineralizers	130	300	X	-	X	1.0	- /0.05	SC	Processed blowdown	Detect operational problems in blowdown processing system	Determine resin exhaustion and performance
Downstream blowdown mixed-bed demineralizers	130	300	X	-	X	1.0	- /0.05	SC	Processed blowdown	Detect operational problems in the blowdown cleanup system	Determine resin exhaustion and performance

a) X denotes that this method of sampling is used.

b) B = boric acid/boron measurement
CC = cation conductivity measurement
SC = specific conductivity measurement
pH = pH measurement
R = online radioactivity measurement
Na = sodium concentration measurement
Cl = chloride concentration measurement
H_2 = hydrogen
O_2 = oxygen

c) Operating temperature >120°F.

TABLE 9.3-5
POTENTIALLY RADIOACTIVE LIQUID STORAGE TANKS

(Sheet 1 of 4)

Tank	System	Number Supplied (Two Units)	Capacity	Location and Elevation	Leak Detection and/or Overflow Detection	Overflow or Relief Valve Discharge Receiving Point	Remarks
Pressurizer relief tank	RCS	2	1800 ft^3	Containment, 822 ft 9 in.	Level indication (CR) high and low alarms (CR)	Via Containment sump to FDT (Alternate - WHT)	Relief through rupture discs
Chiller surge tank fluid	CVCS	2	500 gal	Auxiliary Building, 852 ft 6 in.	Level indication (CR) High alarm	To component cooling water drain tank	Possibility of radio-active entering through a letdown chiller heat exchanger tube leak
Volume control tank	CVCS	2	400 ft^3	Safeguards Building, 831 ft 6 in.	Level indication (CR) high-high, high, low and low-low alarms (CR)	Relief to RHT	_____
Accumulator	SIS	8	1350 ft^3	Containment, 832 ft 6 in. and 842 ft 0 in.	Level indication (CR) high and low alarms (CR)	_____	Assumes RC backflow through check valves
Floor drain tank	WPS	3	a. 10,000 gal b. 10,000 gal c. 30,000 gal	a. Safeguards Building Unit No. 1 b. Safeguards Building Unit No. 2 c. Auxiliary Building	Level indication (LWPP) high and low alarms (CR, LWPP)	Overflow to floor	Tank located within water-tight compartment; NC valve in floor drain line for FDT's 1 & 2 FDT-3 Floods to FDT #1 via FD Sump #4
Laundry and hot shower tank	WPS	1	10,000 gal	Auxiliary Building, 790 ft 6 in.	Level indication (LWPP) high and low alarms (CR, LWPP)	Overflows FD Sump #7 to FDT	_____

TABLE 9.3-5
POTENTIALLY RADIOACTIVE LIQUID STORAGE TANKS

(Sheet 2 of 4)

Tank	System	Number Supplied (Two Units)	Capacity	Location and Elevation	Leak Detection and/or Overflow Detection	Overflow or Relief Valve Discharge Receiving Point	Remarks
Reactor coolant drain tank	WPS	2	350 gal	Containment, 808 ft 0 in.	Level indication (LWPP) high and low alarms (CR, LWPP)	Relief to WHT or FDT via the containment sump	_____
Spent resin storage tank	WPS	1	4100 gal	Auxiliary Building, 810 ft 6 in.	Level indication (LWPP, DP) high and low alarms (CR, LWPP)	Relief to FDT via FD Sump #4	_____
Waste evaporator condensate tank	WPS	1	5000 gal	Auxiliary Building, 790 ft 6 in.	Level indication (LWPP) high and low alarms (CR, LWPP)	Overflows to WHT via FD Sump #6	_____
Waste holdup tank	WPS	1	10,000 gal	Auxiliary Building, 790 ft 6 in.	Level indication (LWPP) high-high, high, and low alarms (CR, LWPP)	Overflows on floor	Refer to Subsection 9.3.3.2.3
Waste monitor tank	WPS	2	5000 gal	Auxiliary Building, 790 ft 6 in.	Level indication (LWPP) high and low alarms (CR, LWPP)	Overflows to FDT via FD Sump #5	_____
Component cooling water surge tank	CCWS	2	4600 gal	Auxiliary Building 873 ft 6 in.	Level indication (CR) high-high, high low-low, and low alarms (CR)	Relief to component cooling water drain tank	Assumes a primary component leakage
Component cooling water drain tank	CCWS	2	2300 gal	Safeguards Building, 773 ft 0 in.	Local level indication High alarm (CR)	Overflows to FDT	Assumes a primary component leakage

TABLE 9.3-5
POTENTIALLY RADIOACTIVE LIQUID STORAGE TANKS

(Sheet 3 of 4)

Tank	System	Number Supplied (Two Units)	Capacity	Location and Elevation	Leak Detection and/or Overflow Detection	Overflow or Relief Valve Discharge Receiving Point	Remarks
Reactor makeup water storage tank	RMWS	2	112,000 gal	Outdoor	Level indication (CR, PDP) low alarm (CR, PDP) High alarm (PDP)	Overflows to WHT	Assumes an accidental makeup of radioactive recycled water
Refueling water storage tank	CSS	2	112,000 gal	Outdoor	Level indication (CR, PDP) high, low, and low-low alarms (CR)	Overflows to FDT	_____
Recycle holdup tank	BRS	2	112,000 gal	Auxiliary Building, 790 ft 6 in.	Level indication (BRP) high and low alarms (CR, LWPP)	Overflows to WHT	_____
Steam generator blowdown spent resin storage tank	steam generator blowdown and cleanup system	1	3700 gal	Auxiliary Building, 790 ft 6 in.	Local level indication (DP, SGBP) high and low alarms (CR, SGBP)	Relief Valve discharge to FDT	Assumes a steam generator tube leak
Chemical drain tank	WPS	1	600 gal	Auxiliary Building, 790 ft 6 in.	Level indication (LWPP) high and low alarms (CR, LWPP)	Overflows to FDT via Sump #7	_____
Laundry holdup and monitor tanks	WPS	2	5000 gal	Auxiliary Building 790 ft 6 in.	Level indication high and low alarms (LWPP, CR)	Overflow to FDT via Sump #7	_____
Condensate storage tank	AFW	2	500,000 gal	Outside	Level indication (CR) high-high, low, and low-low alarms (CR)	Overflow to WHT	_____
Waste conditioning tank	WPS	1	2000 gal	Fuel Building 841 ft 0 in.	Level indication high and low alarms (LWPP, CR)	_____	_____

TABLE 9.3-5
POTENTIALLY RADIOACTIVE LIQUID STORAGE TANKS

(Sheet 4 of 4)

Tank	System	Number Supplied (Two Units)	Capacity	Location and Elevation	Leak Detection and/ or Overflow Detection	Overflow or Relief Valve Discharge Receiving Point	Remarks
Hot phase separator	CPS	1	10,000 gal	Fuel Building 860 ft 0 in.	Level indication high and low alarms	Overflow to back-wash recovery system	_____
Plant Effluent Holdup and Monitoring Tanks	WPS	2	30,000 gal	Yard Inside Protected Area 810’	Level indication high, high-high, low, and low-low	Overflow to Fuel Bldg. Sump 2	_____

NOTE:

SIS	-	Safety Injection System
WPS	-	Waste Processing System
CSS	-	Containment Spray System
BRS	-	Boron Recycle System
CPS	-	Condensate Polishing System
RMWS	-	Reactor Makeup Water System
CR	-	Control Room
LWPP	-	liquid waste processing panel
FDT	-	floor drain tank
WHT	-	waste holdup tank
RHT	-	recycle holdup tank
RTP	-	resin transfer panel
DP	-	drumming panel
BRP	-	boron recycle panel
NC	-	normally closed
PDP	-	potable and demineralized water panel

TABLE 9.3-6
CHEMICAL AND VOLUME CONTROL SYSTEM DESIGN PARAMETERS

General	
Seal Water supply flow rate, for four reactor coolant pumps, nominal (gpm)	32
Seal water return flow rate, for four reactor coolant pumps, nominal (gpm)	12
Letdown flow	
Normal (gpm)	75
Maximum (gpm)[(a)]	120
Charging flow (excludes seal water)	
Normal (gpm)	55
Maximum (gpm)	100
Temperature of letdown reactor coolant entering system (°F)	560
Temperature of charging flow directed to Reactor Coolant System (°F)	518
Temperature of effluent directed to Boron Recycle System (°F)	115
Centrifugal charging pump bypass flow, each (gpm)	60
Amount of 4 weight percent boric acid solution required to meet cold shutdown requirements shortly after full power operation (gallons)	17,800
Maximum pressurization required for hydrostatic testing of Reactor Coolant System (psig)	3,107
Temperature	Ambient

a) Letdown flow maximum refers to the letdown configuration involving flow through both a 45 gpm and 75 gpm letdown orifice. The letdown flow maximum is 140 gpm.

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 1 of 14)

Positive Displacement Pump	
Number	1
Design pressure (psig)	3200
Design temperature (°F)	250
Design flow (gpm)	98
Design head (ft)	5800
Material	Austenitic Stainless Steel
Maximum operating pressure, for Reactor Coolant System hydrotest purposes (psig)	3125
Centrifugal Charging Pumps	
Number	2
Design pressure (psig)	2800
Design temperature (°F)	300
Design flow (gpm)	150
Design head (ft)	5800
Material	Austenitic Stainless Steel
Boric Acid Transfer Pump	
Number	2
Design pressure (psig)	150
Design temperature (°F)	250

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 2 of 14)

Design flow (gpm)	75
Design head (ft)	235
Material	Austenitic Stainless Steel
<u>Chiller Pumps</u>	
Number	2
Design pressure (psig)	200
Design temperature (°F)	200
Design flow (gpm)	400
Design head (ft)	150
Material	Carbon Steel
<u>Regenerative Heat Exchanger</u>	
Number	1
Heat transfer rate at design conditions (Btu/hr)	11.0×10^6
<u>Shell Side</u>	
Design pressure (psig)	2485
Design temperature (°F)	650
Fluid	Borated Reactor Coolant
Material	Austenitic Stainless Steel

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 3 of 14)

Tube Side	
Design pressure (psig)	3100
Design temperature (°F)	650
Fluid	Borated Reactor Coolant
Material	Austenitic Stainless Steel
Shell Side (Letdown)	
Flow (lb/hr)	37,300
Inlet temperature (°F)	560
Outlet temperature (°F)	290
Tube Side (Charging)	
Flow (lb/hr)	27,400
Inlet temperature (°F)	130
Outlet temperature (°F)	518
Letdown Heat Exchanger	
Number	1
Heat transfer rate at design conditions (Btu/hr)	16.1×10^6

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 4 of 14)

Shell Side	
Design pressure (psig)	165
Design temperature (°F)	250
Fluid	Component Cooling Water
Material	Carbon Steel
Tube Side	
Design pressure (psig)	600
Design temperature (°F)	400
Fluid	Borated Reactor Coolant
Material	Austenitic Stainless Steel

Shell Side	Heatup	Normal
Flow (lb/hr)	498,000	170,000
Inlet temperature (°F)	105	105
Outlet temperature (°F)	137	143
Tube Side (Letdown)		
Flow (lb/hr)	59,600	37,300
Inlet temperature (°F)	380	290
Outlet temperature (°F)	115	115

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 5 of 14)

Excess Letdown Heat Exchanger	
Number	1
Heat transfer rate at design conditions (Btu/hr)	5.2×10^6

	Shell Side	Tube Side
Design pressure (psig)	165	2485
Design temperature (°F)	250	650
Design flow (lb/hr)	129,000	12,410
Inlet temperature (°F)	105	560
Outlet temperature (°F)	145	165
Fluid	Component Cooling Water	Borated Reactor Coolant
Material	Carbon Steel	Austenitic Stainless Steel

Seal Water Heat Exchanger	
Number	1
Heat transfer rate at design conditions (Btu/hr)	2.4×10^6

	Shell Side	Tube Side
Design pressure (psig)	165	150
Design temperature (°F)	250	250
Design flow (lb/hr)	186,000	42,200

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 6 of 14)

Inlet temperature (°F)	105	172
Outlet temperature (°F)	118	115
Fluid	Component Cooling Water	Borated Reactor Coolant
Material	Carbon Steel	Austenitic Stainless Steel
<u>Moderating Heat Exchanger</u>		
Number		1
Heat transfer rate at design conditions (Btu/hr)	300	2.53×10^6
Design pressure (psig)	300	300
Design temperature (°F)	200	200
Design flow (lb/hr)	59,600	59,600
Design inlet temperature, boron storage mode (°F)	92.4	72.6
Design outlet temperature, boron storage mode (°F)	140	115
Outlet temperature, boron release mode (°F)	123.7	131.3
Material	Austenitic Stainless Steel	Austenitic Stainless Steel

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 7 of 14)

Letdown Chiller Heat Exchanger

Number	1
Heat transfer rate at design conditions, boron storage mode (Btu/hr)	1.65×10^6

	Shell Side	Tube Side
Design pressure (psig)	150	300
Design temperature (°F)	200	200
Design flow, boron storage mode (lb/hr)	175,000	59,600
Design inlet temperature, boron storage mode (°F)	39	72.6
Design outlet temperature, boron storage mode (°F)	48.4	45
Flow, boron release mode (lb/hr)	175,000	59,600
Inlet temperature, boron release mode (°F)	90	123.7
Outlet temperature, boron release mode (°F)	99.4	96.1
Material	Carbon Steel	Austenitic Stainless Steel

Letdown Reheat Heat Exchanger

Number	1
Heat transfer rate at design conditions (Btu/hr)	1.49×10^6

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 8 of 14)

	Shell Side	Tube Side
Design pressure (psig)	300	600
Design temperature (°F)	200	400
Design flow (lb/hr)	59,600	44,700
Inlet temperature (°F)	115	280
Outlet temperature (°F)	140	246.7
Material	Austenitic Stainless Steel	Austenitic Stainless Steel
Volume Control Tank		
Number		1
Volume (ft^3)		400
Design pressure (psig)		75
Design temperature (°F)		250
Material		Austenitic Stainless Steel
Boric Acid Tanks		
Number		2 shared
Capacity (gal)		46,000
Design pressure (psig)		Atm.
Design temperature (°F)		200
Material		Austenitic Stainless Steel

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT
DATA SUMMARY

(Sheet 9 of 14)

Batching Tank	
Number	1
Capacity (gal)	800
Design pressure (psig)	Atmospheric
Design temperature (°F)	300
Material	Austenitic Stainless Steel
Chemical Mixing Tank	
Number	1
Capacity (gal)	5
Design pressure (psig)	150
Design temperature (°F)	200
Material	Austenitic Stainless Steel
Chiller Surge Tank	
Number	1
Volume (gal)	500
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Carbon Steel

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT
DATA SUMMARY

(Sheet 10 of 14)

Discharge Dampener Tank	
Number	1
Volume, ft^3	2.4
Design pressure, psig Internal	2735
Operating pressure, psig Internal	2350
Design temperature, °F	200
Operating temperature, °F	130
Material	Austenitic Stainless Steel
Suction Stabilizer Tank	
Number	1
Volume, ft^3	2
Design pressure, psig Internal	240
Operating pressure, psig Internal	25
Design temperature, °F	200
Operating temperature, °F	115
Material	Austenitic Stainless Steel

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT
DATA SUMMARY

(Sheet 11 of 14)

Mixed Bed Demineralizers	
Number	1
Design pressure (psig)	300
Design temperature (°F)	250
Design flow (gpm) (a)	120
Resin volume, each (ft^3)	30
Material	Austenitic Stainless Steel

(a) Maximum allowed flow is 170 gpm

Cation Bed Demineralizers	
Number	1
Design pressure (psig)	300
Design temperature (°F)	250
Design flow (gpm) (a)	120
Resin volume (ft^3)	30
Material	Austenitic Stainless Steel

(a) Maximum allowed flow is 170 gpm

Thermal Regeneration Demineralizers	
Number	5

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 12 of 14)

Design pressure (psig)	300
Design temperature (°F)	250
Design flow (gpm)	250
Resin volume (ft^3)	74.3 (Thermal regeneration mode)
Material	Austenitic Stainless Steel

Boric Acid Filter	
Number	1 per unit
Design Pressure (psig)	300
Design temperature (°F)	250
Design Flow (gpm)	250
Micron Rating (M)	0.1 to 40 absolute (100% retention)
Material, vessel	Austenitic Stainless Steel

Letdown Orifice	45 gpm	75 gpm
Number	1	2
Design flow (lb/hr)	22,400	37,300
Differential pressure at design flow (psig)	1700	1700
Design pressure (psig)	2,485	2,485
Design temperature (°F)	650	650

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT
DATA SUMMARY

(Sheet 13 of 14)

Material	Austenitic Stainless Steel	Austenitic Stainless Steel
<u>No. 1 Seal Bypass Orifices</u>		
Number	1/Loop	
Design flow (gpm)	1	
Differential pressure at design flow (psid)	300	
Design pressure (psig)	2485	
<u>Reactor Coolant Filter</u>		
Number	1	
Design pressure (psig)	300	
Design temperature (°F)	250	
Design flow (gpm)	250	
Micron Rating (M)	0.05 to 40 absolute (100% retention)	
Material, vessel	Austenitic Stainless Steel	
<u>Seal Water Injection Filters</u>		
Number	2	
Design pressure (psig)	3100	
Design temperature (°F)	250	
Design flow (gpm)	80	

TABLE 9.3-7
CHEMICAL AND VOLUME CONTROL SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 14 of 14)

Micron Rating (M)	0.45 to 23 absolute (100% retention)	
Material, Vessel	Austenitic Stainless Steel	
Seal Water Return Filter		
Number	1	
Design pressure (psig)	300	
Design Temperature (°F)	250	
Design flow (gpm)	250	
Micron Rating (M)	0.1 to 40 absolute (100% retention)	
Material, vessel	Austenitic Stainless Steel	
Design temperature (°F)		250
Material		Austenitic Stainless Steel
Chiller		
Number		1
Capacity (Btu/hr)		1.66×10^6
Design flow (gpm)		352
Inlet temperature (°F)		48.3
Outlet temperature (°F)		39

TABLE 9.3-8
BORON RECYCLE SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 1 of 4)

Recycle Evaporator Feed Pumps	
Number	2
Design pressure (psig)	150
Design temperature (°F)	250
Design flow (gpm)	30/100
Design head (ft)	320/250
Material	Stainless Steel
Recycle Holdup Tanks	
Number	2 shared
Capacity	112,000
Design pressure (psig)	Atm.
Design temperature (°F)	200
Material	Stainless Steel
Recycle Evaporator Reagent Tank	
Number	1
Capacity (gal)	5
Design pressure (psig)	150
Design temperature (°F)	200
Material	Stainless Steel

TABLE 9.3-8
BORON RECYCLE SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 2 of 4)

Recycle Evaporator Feed Demineralizers	
Number	2
Design pressure (psig)	300
Design temperature (°F)	250
Design flow (gpm)	120
Resin volume (ft^3)	30
Material	Stainless Steel
Recycle Evaporator Condensate Demineralizer	
Number	1
Design pressure (psig)	300
Design temperature (°F)	250
Design Flow (gpm)	120
Resin volume (ft^3)	30
Material	Stainless Steel
Recycle Evaporator Feed Filter	
Number	1/unit
Design pressure (psig)	300
Design temperature (°F)	250
Design flow (gpm)	250
Micron rating (μM)	1 to 20 absolute (100% retention)
Material, vessel	Stainless steel

TABLE 9.3-8
BORON RECYCLE SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 3 of 4)

Recycle Evaporator Condensate Filter	
Number	1 shared
Design pressure (psig)	200
Design temperature (°F)	250
Design flow (gpm)	35
Micron rating (μM)	2 to 40 absolute (100% retention)
Material, vessel	Stainless steel
Recycle Evaporator Concentrates Filter	
Number	1
Design pressure (psig)	200
Design temperature (°F)	250
Design flow (gpm)	35
Micron rating (μM)	2 to 40 absolute (100% retention)
Material, vessel	Stainless steel
Recycle Evaporator Package	
Number	1 shared
Design flow (gpm)	15
Concentration of concentrate boric acid (wt percent)	4
Concentration of condensate	<10 ppm boron as H_3BO_3
Material	Stainless Steel

TABLE 9.3-8
BORON RECYCLE SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

(Sheet 4 of 4)

Recycle Holdup Tank Vent Ejector	
Number	1 shared
Design pressure (psig)	150
Design temperature (°F)	200
Suction flow (scfm)	1
Motive flow (scfm)	40
Material	Carbon Steel

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 1 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
1. Air diaphragm operated globe valve 1-LCV-459 (1-LCV-460 analogous)	a. Fails open	a. Charging and Volume Control - letdown flow.	a. Failure reduces redundancy of providing letdown flow isolation to protect PRZ heaters from uncovering at low water level in PRZ. No effect on system operation. Alternate isolation valve (1-LCV-460) provides backup letdown flow isolation	a. Valve position indication (open to closed position change) at CB.	a. Valve is designed to fail “closed” and is electrically wired so that the electrical solenoid of the air diaphragm operator is energized to open the valve. Solenoid is de-energized to close the valve upon the generation of a SI ”S” signal or a low level PRZ control signal. The valve is electrically interlocked with the level letdown orifice isolation valves and may not be opened manually from the CB if any of these valves are at an open position.
	b. Fails Closed	b. Charging and Volume Control - letdown flow	b. Failure blocks normal letdown flow to VCT. Minimum letdown flow requirements for boration of RCS to hot standby concentration level may be met by establishing letdown flow through alternate excess letdown flow path If the alternate letdown flow path to VCT is not available due to common mode failure (loss of instrument air supply) affecting the opening operation of isolation valves in each flow path, the plant operator can borate the RCS to a hot standby concentration level without letdown flow by taking advantage of the steam space available in the PRZ.	b. Valve position indication (closed to open position change) at CB; letdown flow temperature indication (TI-127) at CV; letdown flow pressure indication (PI-131) at CB; and VCT level indication (LI-112A) and low level alarm at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 2 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
2. Air Diaphragm operated globe valve 1--8149A 1-8149B and 1-8149C analogous	a. Fails Open	a. Charging and Volume Control - letdown flow	a. Failure prevents isolation of normal letdown flow through regenerative heat exchanger when bringing the reactor to a cold shutdown condition after the RHRS is placed into operation. No effect on hot standby operation. Containment isolation (1-8152) may be remotely closed from the CB to isolate letdown flow through the heat exchanger.	a. Valve position indication (open to closed position change) at CB.	1. Valve is of similar design as that stated for item #1. Solenoid is de-energized to close the valve upon the generation of a low level PRZ signal or closing of letdown isolation valves (1-459 and 1-460) upstream of the regenerative heat exchanger.
	b. Fails closed	a. Charging and Volume Control - letdown flow.	b. Failure blocks normal letdown flow to VCT. Minimum letdown flow requirements for boration of RCS to hot standby concentration level may be met by opening letdown orifice isolation valves 1-LCV-8149B and 1-LCV-8849C. If common mode failure (loss of instrument air) prevents opening of these valves and also prevents establishing alternate flow through path, plant operator can borate the RCS to a hot standby concentration level without letdown flow by taking advantage of steam space available in PRZ.	b. Same methods of detection as those stated for item #1, failure made “Fails closed”	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 3 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
3. Air diaphragm operated globe valve 1-8152 (1-8160 analogous)	a. Fails closed	a. Charging and Volume Control - letdown flow	a. Same effect on system operation as that stated for item #1, failure mode “Fails closed”.	a. Same methods of detection as those stated for item #1, failure mode “fails closed”. In addition, close position group monitoring light at CB.	1. Valve is of similar design as that stated for item #1. Solenoid is de-energized to close the valve upon the generation of an ESF “T” signal.
	b. Fails open.	b. Charging and Volume Control - letdown flow	b. Failure has no effect on CVCS operation during normal plant operation and load follow. However, under accident conditions requiring containment isolation, failure reduces the redundancy of providing isolation of normal letdown line.	b. Valve position indication (open to closed position change) at CB.	
4. Air diaphragm operated globe valve 1-TCV-381B	a. Fails open	a. Boron Concentration Control - boron thermal regeneration (boration)	a. Failure inhibits use of BTRS for load follow operation (boration) due to low temperature of letdown flow entering BTRS demineralizers. Alternate boration of reactor coolant for load follow is possible using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	a. Letdown heat exchanger tube discharge flow (FI-132) and pressure (PI-131) indications at CB and BTR demineralizer inlet flow temperature indication (TI-381) at CB if BTRS is in operation.	1. Valve is designed to fail “open” and is electrically wired so that the electrical solenoid of the air diaphragm operator is energized to close the valve.
	b. Fails closed	b. Boron Concentration Control - boron thermal regeneration (boration)	b. Failure inhibits use of BTRS for load follow operation (boration) due to loss of temperature control of letdown flow entering BTRS demineralizers.	b. Same method of detection as those stated for item #1, failure mode “Fails closed” except no “closed to open position change” indication at CB.	2. BTRS operation is not required in operations of CVCS systems used to bring the reactor to hot standby condition.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 4 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
			Failure also blocks normal letdown flow to VCT when BTRS is not being used for load follow. Minimum letdown flow requirements for boration of RCS to hot standby concentration level may be met as stated for effect on system operation for item #1, failure mode “fails closed”.		
5. Air diaphragm operated globe valve 1-PCV-131	a. Fails open.	a. Charging and Volume Control - letdown flow	a. Failure prevents control of pressure to prevent flashing of letdown flow in letdown heat exchanger and also allows high pressure fluid to demineralizers. Relief valve (1-8119) open in demineralizer line to release pressure to VCT and valve (1-TCV-129) changes position to divert flow to VCT. Boration of RCS to hot standby concentration level is possible with valve failing open.	a. Letdown heat exchanger tube discharge flow indication (FI-132) and high flow alarm at CB; temperature indication (TI-130) and high temperature alarm at CB; and pressure indication (PI-131) at CB.	1. Same remark as stated for item #4, in regards to valve design. 2. As a design transient the letdown heat exchanger is designed for complete loss of charging flow.
	b. Fails Closed.	b. Charging and Volume Control -letdown flow.	b. Same effect on system operation as that for item #1, failure mode “fails closed”.	b. Letdown heat exchanger discharge flow indication (FI-132), and pressure indication (PI-131) and high pressure alarm at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 5 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
6. Air diaphragm operated globe three way valve 1-TCV-129	a. Fails open for flow only to VCT.	a. Charging and Volume Control - letdown flow.	a. Letdown flow bypassed from flowing to demineralizers and BTRS. Failure prevents ionic purification of letdown flow and inhibits operation of BTRS. Boration of RCS to hot standby concentration level is possible with valve failing open for flow only to VCT.	a. Valve position indication (VC Tank) at CB and RCS activity level when sampling letdown flow.	1. Electrical solenoid of air diaphragm operator is electrically wired so that solenoid is energized to open valve for flow to the demineralizers. Valve opens for flow to VCT on “High Letdown Temperature” or on “High Letdown Reheat HX Outlet Temperature”.
	b. Fails open for flow only to demineralizer.	b. Charging and Volume Control - letdown flow.	b. Continuous letdown to demineralizers and BTRS. Failure prevents automatic isolation of demineralizers and BTRS under fault condition of high letdown flow temperatures. These systems may be manually isolated using local valves Boration of RCS to hot standby concentration level is possible with valve failing open for flow only to demineralizer.	b. Valve position indication (Demin.) at CB. If BTRS is in operation, BTR demineralizer return flow indiction (FI-385) indicating flow during an alarm condition of high letdown reheat. Heat exchanger outlet temperature or high letdown temperature.	2. Technical Specifications provide a limit on RCS activity.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 6 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
7. Air diaphragm operated diaphragm valve 1-7054	a. Fails closed.	a. Boron Concentration Control - boron thermal regeneration or storage	a. Failure inhibits use of BTRS for load follow operation (Boration or dilution) due to flow isolation of the BTRS. Alternate boration or dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operation to bring reactor to hot standby condition.	a. Valve position indication (closed to open position change) at CB: BTRS operation indication (borate or dilute) at CB and BTR demineralizer return flow indication (FI-385) and inlet flow temperature indication (TI-381) at CB.	1. Valve is designed to fail “closed” and is electrically wired so that the electric solenoid of air diaphragm operator is energized to the open valve. 2. BTRS not required to bring reactor to hot standby condition.
8. Air diaphragm operated diaphragm valve 1-7002A	a. Fails closed.	a. Boron Concentration Control - boron storage	a. Failure inhibits use of BTRS for load follow operation (dilution) due to follow isolation of letdown chiller heat exchanger. Alternate dilution of reactor coolant for load follow may be accomplished using RMCV of CVCS. No effect on operations to bring reactor to hot standby condition.	a. BTRS operation indication (dilute) at CB; letdown reheat heat exchanger outlet temperature (TI-381) at CB; and RCS boron level when sampling letdown flow.	1. Same remarks as those stated above for item #7.
	b. Fails open	b. Boron Concentration Control - boron thermal regeneration	b. Failure inhibits use of BTRS for load follow operation (boration) due to flow through letdown chiller heat exchanger. Alternate boration of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operation to bring reactor to hot standby condition.	b. BTRS operation indication (boration) at CB: BTRS return flow temperature indication (TI-386) at CB; BTR return flow indication (FI-385) at CB; and RCS boron level when sampling letdown flow.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 7 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
9. Air diaphragm operated diaphragm valve 1-7002B	a. Fails closed.	a. Boron Concentration Control boron storage	a. Same effect on system operation as that stated above for item #8, failure mode “Fails closed”.	a. Same methods of detection as those stated above for item #8 failure mode “Fails closed”.	1. Same remark as those stated above for item #7.
	b. Fails open.	b. Boron Concentration Control - boron thermal regeneration	b. Failure inhibits use of BTRS for load follow operation (boration) due to bypass of letdown flow from letdown reheat heat exchanger. Alternate boration of reactor coolant may be accomplished using RMCS of CVCS. No effect on operation to bring reactor to hot standby condition.	b. Same methods of detection as those stated above for item #8, failure mode “Fails open”.	
10. Relief Valve 1-8117	a. Fails open.	a. Charging and Volume Control - letdown flow	a. Letdown flow is relieved to pressurizer relief tank. Failure inhibits use of demineralizers for reactor coolant purification and use of BTRS. Normal letdown line can be isolated and minimum letdown flow requirements for hot standby may be met by establishing letdown flow through alternate excess letdown flow path.	a. High temperature relief line indication (TI-125) and alarm at CB and VCT level indication (LI-112A) and low level alarm at CB.	1. Radioactive fluid contained.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 8 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
11. Relief Valve 1-1889	a. Fails open.	a. Charging and Volume Control - letdown flow.	a. Letdown flow is relieved to VCT. Failure inhibits use of demineralizers for reactor coolant purification and use of BTRS. Normal letdown line can be isolated and minimum letdown flow requirement for hot standby may be met by establishing flow through alternate excess letdown flow path.	a. RCS activity level when sampling letdown flow. When BTRS is operating, low BTR demineralizer return flow indication (FI-385) at CB.	1. Radioactive fluid contained.
12. Air diaphragm operated diaphragm valve 1-8245	a. Fails closed.	a. Boron Concentration Control - boron thermal regeneration or storage	a. Purification of reactor coolant using only mixed bed demineralizers cannot be performed. Failure also blocks normal letdown flow. Boration of RCS to hot standby concentration level may be met as stated for effect on system operation for item #1, failure mode “Fails closed”.	a. BTRS operation indication (off) at CB and RCS activity level when sampling letdown flow. Valve position indication (closed to open position change) at CB.	1. Same remark as that stated for item #4.
	b. Fails open.	b. Boron Concentration Control - boron thermal regeneration or storage.	b. Failure inhibits use of BTRS for load follow operation (boration or dilution) due to bypass of letdown flow from BTRS. Alternate boration or dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	b. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (borate or dilute) at CB and low BTR demineralizer return flow indication (FI-385) at CB. Valve position indication (open to closed position change) at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 9 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
13 Air diaphragm operated diaphragm valve 1-7056 (1-7045 analogous)	a. Fails closed.	a. Boron Concentration Control - boron storage.	a. Failure inhibits use of BTRS for load follow operation (dilution) due to flow isolation of BTR demineralizers. Alternate dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	a. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (dilute) at CB and low BTR demineralizer return flow indication (FI-385) at CB.	1. Same remarks as those stated for item #7.
	b. Fails open.	b. Boron Concentration Control - boron thermal regeneration.	b. Failure inhibits use of BTRS for load follow operation (boration) due to flow bypass of BTR demineralizers. Alternate boration of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	b. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (borate) at CB.	
14. Air diaphragm operated diaphragm valve 1-7057	a. Fails open.	a. Boron Concentration Control - boron storage	a. Failure inhibits use of BTRS for load follow operation (dilution) due to flow bypass of BTR Demineralizers. Alternate dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	a. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (dilute) at CB.	1. Same remark as that stated for item #4.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 10 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
	b. Fails closed.	b. Boron Concentration Control - boron thermal regeneration.	b. Failure inhibits use of BTRS for load follow operation (boration) due to flow isolation of BTR demineralizers. Alternate boration of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	a. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (borate) at CB and low BTR demineralizer return flow indication (FI-385) at CB.	
15. Air diaphragm operated diaphragm valve 1-7040	a. Fails open.	b. Boron Concentration Control - boron thermal regeneration.	a. Same effect on system operation as that stated for item #14, failure mode “Fails open”.	a. Same methods of detection as those stated for item #14, failure mode “Fails open”.	
	b. Fails closed.	b. Boron Concentration Control - boron thermal regeneration.	b. Failure inhibits use of BTRS for load follow operation (boration) due to blockage of return letdown flow from letdown chiller heat exchanger. Alternate boration of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	b. Same methods of detection as those stated above for item #14, failure mode, “Fails closed”.	1. Same remark as that stated for item #4.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 11 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
16. Air diaphragm operated diaphragm valve 1-7041	a. Fails open.	a. Boron Concentration Control - boron storage	a. Failure inhibits use of BTRS for load follow operation (dilution) due to flow bypass of letdown chiller heat exchanger. Alternate dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	a. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (dilute) at CB and letdown reheat heat exchanger outlet temperature indication (TI-381) at CB.	1. Same remark as that stated for item #4.
	b. Fails closed.	b. Boron Concentration Control - boron thermal regeneration.	b. Failure inhibits use of BTRS for load follow operation (boration) due to flow isolation of letdown reheat heat exchanger and BTR demineralizers. Alternate boration of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operation to bring reactor to hot standby condition.	b. Same methods of detection as those stated for item #14, failure mode “Fails closed”.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 12 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
17. Air diaphragm operated diaphragm valve 1-7022	a. Fails closed.	a. Boron Concentration Control - boron storage.	a. Failure inhibits use of BTRS for load follow operation (dilution) due to flow blockage of return letdown flow from moderating heat exchanger. Alternate dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	a. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (dilute) at CB and low BTR demineralizer return flow indication (FI-385) at CB.	1. Same remarks as those stated for item #7.
	b. Fails open.	b. Boron Concentration Control - boron thermal regeneration.	b. Failure inhibits use of BTRS for load follow operation (boration) due to bypass of flow from letdown chiller heat exchanger of return letdown flow. Alternate boration of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operation to bring reactor to hot standby condition.	b. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operate indication (borate) at CB and BTRS return flow temperature indication (TI-386) and high temperature alarm at CB.	
18. Air diaphragm operated butterfly valve 1-TCV-386	a. Fails closed.	a. Boron Concentration Control - boron thermal regeneration and storage.	a. Failure inhibits use of BTRS for load follow operation (boration and dilution) due to flow blockage of chiller flow through letdown chiller heat exchanger. Alternate boration and dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operation to bring reactor to hot standby condition.	a. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operate indication (borate or dilute) at CB; BTRS return flow temperature indication (TI-386) and high temperature alarm at CB; and chiller surge tank temperature indication (TI-379) at CB.	1. Valve is designed to fail “closed”. 2. BTRS not used to bring the reactor to hot standby condition.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 13 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
19. Air diaphragm operated butterfly valve 1-FCV-375.	a. Fails open.	a. Boron Concentration Control - boron thermal regeneration and storage	a. Failure inhibits use of BTRS for load follow operation (boration and dilution) due to flow bypass of chiller flow from letdown chiller heat exchanger. Alternate boration and dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	a. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS return flow temperature indication (TI-386) and high temperature alarm at CB and chiller surge tank temperature indication (TI-379) at CB.	1. Valve is designed to fail “open”. 2. BTRS not used to bring the reactor to hot standby condition.
20. Chiller Unit AHCU	a. Fails to cool liquid	a. Boron Concentration Control - boron thermal regeneration and storage.	a. Failure inhibits use of BTRS for load follow operation (boration and dilution) due to loss of cooling capability of let-down chiller heat exchanger. Alternate boration and dilution of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operations to bring reactor to hot standby condition.	a. Same methods of detection as those stated above for item #19. In addition, BTRS operation indication (borate or dilute) at CB.	1. BTRS not used to bring the reactor to hot standby condition.
21. Chiller Pump #1 APCI-1 (Chiller Pump #2 analogous)	a. Fails to deliver working fluid.	a. Boron Concentration Control - boron thermal regeneration and storage.	a. No effect on BTRS system operation. Redundant chiller pump #2 provides necessary delivery of working fluid for chiller unit operation. BTRS not required in operations to bring reactor to hot standby condition.	a. BTRS operation indication (borate or dilute) at CB and local pump discharge flow pressure indication (PI-377A).	1. Both chiller pumps operate simultaneously during BTRS system operation.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 14 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
22. Air diaphragm operated globe valve 1-TCV-381A	a. Fails closed.	a. Boron Concentration Control - boron thermal regeneration.	a. Failure inhibits use of BTRS for load follow operation (boration) due to flow isolation of shell side of letdown reheat heat exchanger. Alternate boration of reactor coolant for load follow may be accomplished using RMCS of CVCS. No effect on operation to bring reactor to hot standby condition.	a. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (borate) at CB and letdown reheat heat exchanger outlet temperature indication (TI-381) at CB.	1. Same remarks as those stated for item #7.
	b. Fails open	b. Boron Concentration Control - boron storage.	b. Failure inhibits use of BTRS for load follow operation (dilution) due to passage of CVCS letdown flow through tube side of letdown reheat heat exchanger. Alternate dilution of reactor coolant may be accomplished using RMCS of CVCS. No effect on operation to bring reactor to hot standby condition.	b. RCS boron level when sampling letdown flow. If BTRS is operating, BTRS operation indication (dilute) at CB and letdown reheat heat exchanger outlet temperature indication (TI-381) at CB.	
23. Air diaphragm operated globe valve 1-8153 (1-8154 analogous)	a. Fails closed.	a. Charging and Volume Control - letdown flow.	a. Failure inhibits use of the excess letdown fluid system of the CVCS as an alternate system that may be used for letdown flow control during normal plant operation and inhibits use of the excess letdown system to control water level in the pressurizer of the RCS during final stage of plant startup due to flow blockage.	a. Valve position indication (closed to open position change) at CB and excess letdown heat exchanger outlet pressure indication (PI-124) and temperature indication (TI-122) at CB.	1. Same remark as that stated for item #7 in regards to valve design. 2. If normal letdown and excess letdown flow is not available for hot standby operation, plant operator can borate RCS to hot standby concentration using steam space available in PRZ.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 15 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
	b. Fails open.	b. Charging and Volume Control - letdown flow.	b. Failure reduces redundancy of providing excess letdown flow isolation during normal plant operation and for plant startup. No effect on system operation. Alternate isolation valve (1-1854) closes to provide backup flow isolation of excess letdown line.	b. Valve position indication (open to closed position change) at CB.	
24. Air diaphragm operated globe valve 1-HCV-123	a. Fails closed.	a. Charging and Volume Control - letdown flow.	a. Same effect on system operation as stated for item #23, failure mode “Fails closed”.	a. Same methods of detection as those stated for item #23, failure mode “Fails closed” except for valve position indication at CB.	1. Same remarks as those stated above for item #23.
	b. Fails open.	b. Charging and Volume Control - letdown flow.	b. Failure prevents manual adjustment at CB of RCS system pressure downstream of excess letdown heat exchanger to a low pressure consistent with No. 1 seal leakoff back-pressure requirements. When using excess letdown system failure leads to a decrease in seal water pump shaft flow for cooling pump bearings.	b. Excess letdown heat exchanger outlet pressure indication (PI-124) at CB, and seal water return flow recording (FR-157) and low flow alarm at CB.	
25. Air diaphragm operated diaphgram valve 1-LCV-181 (1-LCV-178, 1-LCV-179, and 1-LCV-180 analogous)	a. Fails closed.	a. Charging and Volume Control - seal water flow.	a. No automatic makeup of seal water to seal standpipe that services No. 3 seal of RC pump #1. No effect on operation to bring the plant to hot standby condition.	a. Valve position indication (closed to open position change) and low standpipe level alarm at CB.	1. Same remark as that stated for item #7 in regards to valve design.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

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Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
	b. Fails open.	b. Charging and Volume Control - seal water flow.	b. Overfill of seal water standpipe and dumping of reactor makeup water to containment sump during automatic makeup of water for No. 3 seal of RC pump #1. No effect on operations to bring reactor to hot standby condition.	b. Valve position indication (open to closed position change) and high standpipe level alarm at CB.	2. Low level standpipe alarm conservatively set to allow additional time for RC pump operation without a complete loss of seal water being injected to No. 3 seal after sounding of alarm.
26. Relief valve 1-8121 2CS-8000 (Unit 2)	a. Fails open.	a. Charging and Volume Control - seal water flow.	a. RC pump seal water return flow and excess letdown flow bypassed to PRZ relief tank of RCS. Failure inhibits use of the excess letdown fluid system of the CVCS as an alternate system that may be used for letdown flow control during normal plant operation and inhibits use of excess letdown system to control water level in the PRZ of the RCS during final stage of a plant startup.	a. Decrease in VCT level causing RMCS of CVCS to operate	1. The capacity of the relief valve equals maximum flow from four RC pump seals plus excess letdown flow. 2. Radioactive fluid contained. 3. Same as remark #2 noted for item #23.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

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Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
27. Motor operated globe valve 1-8112 (1-8100 analogous)	a. Fails open.	a. Charging and Volume Control - seal water flow and excess let-down flow.	a. Failure has no effect on CVCS operation during normal plant operation and load follow. However, under accident conditions requiring containment isolation failure reduces redundancy of providing isolation of seal water flow and excess letdown flow.	a. Valve position indication (open to closed position change) at CB.	1. Valve is normally at a full open position and motor operator is energized to close the valve upon the generation of an ESF "T" signal. 2. If normal letdown and excess letdown flow is not available for hot standby operation, plant operator can borate RCS to hot standby concentration using steam space available in PRZ.
	b. Fails closed.	b. Charging and Volume Control - seal water flow and excess let-down flow.	b. RC pump seal water return flow and excess letdown flow blocked. Failure inhibits use of the excess letdown fluid system of the CVCS as an alternate system that may be used for letdown flow control during normal plant operation and degrade cooling capability of seal water in cooling RC pump bearings.	b. Valve position indication (closed to open position change) at CB; group monitoring light and alarm at CB; and seal water return flow recording (FR-157) and low seal water return flow alarm at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 18 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
28. Motor operated gate valve 1-8105 (1-8106 analogous)	a. Fails open.	a. Charging and Volume Control - charging flow.	a. Failure has no effect on CVCS operation during normal plant operation and load follow. However, under accident condition requiring isolation of charging line, failure reduces redundancy of providing isolation of normal charging flow.	a. Valve position indication (open to closed position change) at CB.	1. Valve is normally at a full open position and motor operator is energized to close the valve upon the generation of a Safety Injection “S” signal.
	b. Fails closed.	b. Charging and Volume Control - charging flow.	b. Failure inhibits use of normal charging line to RCS for boration, dilution, and coolant makeup operations. Seal water injection path remains available for boration of RCS to a hot standby concentration level and makeup of coolant during operations to bring the reactor to hot standby condition.	b. Valve position indication (closed to open position change) and group monitoring light (valve closed) at CB; letdown temperature indication (TI-127) and high temperature alarm at CB; charging flow temperature indication (TI-126) at CB; seal water flow pressure indication (PI-120A) at CB; VCT level indication (LI-112A and LI-185) and high level alarm at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 19 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
29. Air diaphragm operated globe valve 1-HCV-182	a. Fails open.	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure prevents manual adjustment at CB of seal water flow through the control of back pressure in charging header resulting in a reduction of flow to RC pump seals leading to a reduction in flow to RCS via labyrinth seals and pump shaft flow for cooling pump bearing. Boration of RCS to a hot standby concentration level and makeup of coolant during operations to bring reactor to hot standby condition is still possible through normal charging low path.	a. Seal water flow pressure indication (PI-120A) at CB; seal water return recording (FR-157); and low seal water return flow alarm at CB.	1. Same remark as that stated for item #4 in regards to design of valve.
	b. Fails closed.	b. Charging and Volume Control - charging flow.	b. Same effect on system operation as that stated for item #28, failure mode “Fails closed”.	b. Same method of detection as those stated above for item #28, failure mode “Failure closed”.	
30. Motor operated globe valve 1-8110 (1-8111 analogous)	a. Fails open.	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure has no effect on CVCS operation during normal plant operation and load follow. However, under accident condition requiring isolation of centrifugal charging pump miniflow line, failure reduces redundancy of providing isolation of mini-flow to suction of pumps via seal water heat exchanger.	a. Valve position indication (open to closed position change) at CB.	1. Same remark as that stated for item #28.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

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Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
	b. Fails closed.	b. Charging and Volume Control - charging flow and seal water flow.	b. Failure blocks miniflow to suction of centrifugal charging pumps via seal water heat exchanger. Normal charging flow and seal water flow prevents deadheading of pumps when used. Boration of RCS to a hot standby concentration level and makeup of coolant during operations to bring reactor to hot standby condition is still possible.	b. Valve position indication (closed to open position change) at CB; group monitoring light (valve closed) and alarm at CB: and charging and seal water flow indication (FI-121A) and high flow alarm at CB.	
31. Air diaphragm operated globe valve 1-8146	a. Fails open	a. Charging and Volume Control - charging flow.	a. Failure has no effect on CVCS operation during normal plant operation. Valve is used during cold shutdown operation to isolate normal charging line when using the auxiliary spray during the cooldown of the pressurizer. Cold shutdown of reactor is still possible, however, time for cooling down RPZ will be extended.	a. Valve position indication (open to closed position change) at CB.	1. Same remark as that stated for item #4 in regards to design of valve.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 21 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
	b. Fails closed.	b. Charging and Volume Control - charging flow.	b. Failure blocks normal charging flow to the RCS. No effect on CVCS operations during normal plant operation, load follow or hot standby operation. Plant operation can maintain charging flow by establishing flow through alternate charging path by opening of isolation valve (1-8147)	a. Valve position indication (closed to open position change) at CB; charging flow indication (TI-126) at CB; regenerative heat exchanger shell side exit temperature indication (TI-127) and high temperature alarm at CB; and charging and seal water flow indication (FI-121A) and low flow alarm at CB.	
32. Air diaphragm operated globe valve 1-8147	a. Fails closed.	a. Charging and Volume Control - charging flow.	a. Failure reduces redundancy of charging flow paths to RCS. No effect on CVCS operations during normal plant operation, load follow, or hot standby operation. Normal charging flow path remains available for charging flow.	a. Valve position indication (closed to open position change) at CB.	1. Same remark as that stated for item #4 in regards to design of valve.
	b. Fails open.	b. Charging and Volume Control - charging flow.	b. Same effect on system operation and shutdown as that stated above for item #31, failure mode “fails open” if alternate charging line is in use.	b. Valve position indication (open to closed position change) at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 22 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
33. Air diaphragm operated globe valve 1-8145	a. Fails open.	a. Charging and Volume Control - charging flow.	a. Failure results in inadvertent operation of auxiliary spray that results in a reduction of PRZ pressure during normal plant operation and load follow. PRZ heaters operate to maintain required PRZ pressure Boration of RCS to a hot standby concentration level and makeup of coolant during operation to bring reactor to hot standby condition is still possible.	a. Valve position indication (open to closed position change) at CB and PRZ pressure recording (PR-455) and low pressure alarm at CB.	1. Same remark as that stated for item #7 in regards to design valve.
	b. Fails closed	b. Charging and Volume Control - charging flow	b. Failure has no effect on CVCS operation during normal plant operation, load follow and hot standby operation. Valve is used during cold shutdown operation to activate spray cooling down the pressurizer after operation of RHRS.	b. Valve position indication (closed to open position change) at CB.	
34. Relief Valve 1-8123	a. Fails open.	a. Charging and Volume Control - charging flow.	a. Failure results in a portion of a seal water return flow and centrifugal charging pump miniflow being bypassed to VCT. Boration of RCS to a hot standby concentration level makeup of coolant during operations to bring reactor to hot standby condition is still possible.	a. Local pressure indication (PI-118 and PI-119) in discharge line of centrifugal charging pumps.	1. Radioactive fluid contained.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 23 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
35. Relief Valve 1-8118	a. Fails open.	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure results in a portion of charging flow and seal water from constant displacement pump being bypassed to VCT. No effect on normal plant operation, load follow or bringing reactor to hot standby condition. Constant displacement pump may be taken out of service and an alternate centrifugal charging pump used for delivery of charging and seal water flow.	a. Local pressure indication (PI-117) in discharge line of constant displacement pump.	1. Constant displacement pump may be flow isolation by closing manual gate valves in discharge and suction lines of pump. 2. Radioactive fluid contained.
36. Air diaphragm operated globe valve -1FCV-121	a. Fails open.	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure reduces redundancy of providing charging and seal water flow to RCS. No effect on normal plant operation, load follow, or bringing reactor to hot standby condition. Constant displacement pump normally used for delivery of charging and seal water flow to RCS. Check valves (1-8481A and 8481B) provide isolation of displacement flow to discharge of centrifugal pump if valve fails “open” during operation of constant displacement pump.	a. Charging and seal water flow indication (FI-121A) and high flow alarm at CB, and PRZ level recording (LR-459) and high level alarm at CB.	1. Same remark as that stated for item #4 in regards to design of valve. 2. Methods of detection apply when a centrifugal charging pump is in operation.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 24 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
	b. Fails closed.	b. Charging and Volume Control - charging flow and seal water flow.	b. Failure reduces redundancy of providing charging and seal water flow to RCS. No effect on system operation during normal plant operation, load follow, or bringing reactor to hot standby condition. Constant displacement pump normally used for delivery of charging and seal water flow to RCS. Valve failing closed under an accident condition requiring flow delivery by centrifugal charging inhibits flow from the pumps.	b. Charging and seal water flow indication (FI-121A) and low flow alarm at CB, and PRZ level recording (LR-459) and low level alarm at CB.	
37. Check valve 1-8497	a. Fails open.	a. Charging and Volume Control - charging flow and and seal water flow.	a. Failure reduces redundancy of providing charging and seal water to RCS. Discharge of constant displacement pump remains open to “back-flow” when a centrifugal charging pump is placed into operation. No effect on normal plant operation, load follow, or bringing reactor to hot standby conditon; constant displacement pump normally used for delivery of charging and seal water flow.	a. Charging and seal water flow indication (FI-121A) and low flow alarm at CB, and PRZ level recording (LR-459) and low level alarm at CB.	1. Constant displacement pump may be isolated by the closing of the manual valves in pump’s suction and discharge lines. 2. Methods of detection apply when centrifugal charging pump #1 is in operation.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 25 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
38. Check valve 1-8481A (1-8481B analogous)	a. Fails open	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure reduces redundancy of providing charging and seal water flow to RCS. Discharge of centrifugal charging pump #1 is open to "back-flow" when centrifugal charging pump #2 is placed into operation after failure of centrifugal charging pump #1 to deliver charging and seal water flow. No effect on normal plant operation, load follow, or bringing reactor to hot standby condition.	a. Same methods of detection as those stated for item #37.	1. Centrifugal charging pump #1 may be isolated by the closing of manual valves in pump's suction and discharge lines.
39. Positive displacement pump APPD	a. Fails to deliver working fluid	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure reduces redundancy of providing charging and seal water flow to RCS. No effect on normal plant operation, load follow, or bringing reactor to hot standby condition. Centrifugal charging pump (#1 or #2) may be placed into operation for delivery of charging and seal water flow.	a. Pump circuit breaker position indication (open) at CB; common pump breaker trip alarm at CB; charging and seal water flow indication (FI-121A) and low flow alarm at CB; and PRZ level recording (LR-459) and low level alarm at CB.	1. Pump stroke is regulated to control amount of charging flow delivered to the PRZ.
40. Centrifugal charging pump #1 APCH-1 (Pump #2 analogous)	a. Fails to deliver working fluid.	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure reduces redundancy of providing charging and seal water flow to RCS from the centrifugal charging pumps. No effect on normal plant operation, load follow, or bringing reactor to hot standby condition.	a. Same methods of detection as those stated above for item #39 when centrifugal charging pump #1 is in operation	1. Flow rate for a centrifugal charging pump is controlled by a modulating valve (1-FCV-121) in discharge header for the centrifugal charging pumps.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 26 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
41. Air diaphragm operated globe valve 1-8156.	a. Fails closed.	a. Chemical Control, Purification and Makeup - oxygen control.	a. Failure blocks hydrogen flow to VCT and leads to loss of venting of VCT (vent valve (1-PCV-115 closes on low VCT pressure) resulting in loss of gas stripping of fission products from RCS coolant. No effect on operation to bring the reactor to hot standby condition.	a. VCT pressure indication (PI-115A) and low pressure alarm at CB. Periodic sampling of gas mixture in VCT.	1. Valve is designed to fail “closed”. 2. Plant’s technical specification set limits on RCS activity level.
42. Relief valve 1-8120	a. Fails open.	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure allows VCT liquid to be relieved to BRS recycle holdup tank resulting in a loss of VCT liquid makeup coolant available for charging and seal water flow during normal plant operation, load follow, and bringing the reactor to a hot standby condition. VCT isolation valves (1-LCV-112B and 1-LCV-112C) close on low-low tank level signal causing the suction of charging pumps to be transferred to the RWST for an alternate supply of borated coolant.	a. Decrease in VCT level causing RMCS to operate; VCT level indications (LI-112A and LI-185) and low level alarms (low and low-low) at CB; and BRS recycle holdup tank level increase.	1. Radioactive fluid contained.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 27 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
43. Motor operated gate valve 1-LCV-112B (1-LCV-112C analogous)	a. Fails open.	a. Charging and Volume Control - charging flow and seal water flow.	a. Failure has no effect on CVCS operation during normal plant operation, load follow, and bringing reactor to a hot standby condition. However, under accident conditions requiring isolation of VCT, failure reduces redundancy of providing isolation for discharge line of VCT.	a. Valve position indication (open to closed position change) at CB.	1. During normal plant operation and load follow valve is at a full open position and the motor operator is energized to close the valve upon the generation of a VCT low-low water level signal or upon the generation of a Safety Injection “S” signal, providing isolation valve 1-LCV-112D (1-LDV-112E analogous) is at a full open position.
	b. Fails closed.	b. Charging and Volume Control - charging flow and seal water flow.	b. Failure blocks fluid flow from VCT during normal plant operation, load follow and when bringing the reactor to a hot standby condition. Alternate supply of borated coolant from the TWST to suction of charging pumps can be established from the CB by the operator through the opening of RWST isolation valves (1-LCV-112D and 1-LCV-112E).	b. Valve position indication (closed to open position change) at CB; group monitoring light (valve closed) at CB; charging and seal water flow indication (FI-121A) and low flow alarm at CB; and RPZ level recording (LR-459) and low level alarm at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 28 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
44. Air diaphragm operated diaphragm valve 1-PCV-115	a. Fails closed.	a. Chemical Control, Purification and Makeup - oxygen control.	a. Failure blocks venting of VCT gas mixture to gas waste processing system (hydrogen recombiners) for stripping of fission products from RCS coolant during normal plant operation and load follow. No effect on operations to bring the reactor to hot standby condition.	a. Valve position indication (closed to open position change) at CB and VCT pressure indication (PI-115A) at CB. Periodic sampling of gas mixture in VCT.	1. Same remark as that stated for item #7 in regards to valve design. 2. Same remark as that stated for item #41 in regards to RCS activity.
45. Air diaphragm operated diaphragm valve 1-FCV-110B	a. Fails closed.	a. Boron Concentration Control - reactor makeup control - boration, auto make-up, and alternate dilution.	a. Failure blocks fluid flow from reactor makeup control system for automatic boric acid addition and reactor water makeup during normal plant operation and load follow. Failure also reduces redundancy of fluid flow paths for dilution of RC coolant by reactor makeup water and blocks fluid flow for boration of the RC coolant when bringing the reactor to a hot standby condition. Boration (at B A tank boron concentration level) of RCS coolant to bring the reactor to hot standby condition is possible by opening of alternate BA tank isolation valve (1-8104) at CB.	a. Valve position indication (closed to open position change) at CB; total makeup flow deviation alarm at CB; and VCT level indications (LI-112A and LI-185) and low level alarms (low and low low) at CB.	1. Same remark as that stated for item #7 in regards to valve design.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 29 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
	b. Fails open	b. Boron Concentration Control - reactor makeup control - boration, auto make-up, and alternate dilution	b. Failure allows for alternate dilute mode type operation for system operation of normal dilution of RCS coolant. No effect on CVCS operation during normal plant operation and load follow, and when bringing the reactor to a hot standby condition.	b. Valve position indication (open to closed change) at CB.	
46. Air diaphragm operated diaphragm valve 1-FCV-111B	a. Fails closed.	a. Boron Concentration Control - reactor makeup control - dilution and alternate dilution.	a. Failure blocks fluid flow from RMCS for dilution of RCS coolant during normal plant operation and load follow. No effect on CVCS operation. Operator can dilute RCS coolant by establishing “alternate dilute” mode of system operation. Dilution of RCS coolant not required when bringing the reactor to a hot standby condition.	a. Same methods of detection as those stated above item #45, failure mode “Fails closed”.	1. Same remark as that stated for item #7 in regards to valve design.
	b. Fails open.	b. Boron Concentration Control - reactor makeup control - dilution and alternate dilution.	b. Failure allows for alternate dilute mode type operation for system operation of boration and auto makeup of RCS cool-ant. No effect on CVCS operation during normal plant operation and load follow and when bringing the reactor to a hot standby condition.	b. Valve position indication (open to closed position change) at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

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Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
47. Relief Valve 1-8124.	a. Fails open.	a. Charging and Volume Control - charging and seal water flow.	a. Failure allow for a portion of flow to suction header of charging pumps to be relieved to BRS recycle holdup tank. Boration of RCS coolant to bring reactor to hot standby condition is still possible.	a. Decrease in VCT level causing RMCS to operate; VCT level indications (KI-112A and LI-185) and low level alarms (low and low-low) at CB; and BRS recycle holdup tank level increase.	1. Radioactive fluid contained.
48. Air diaphragm operated globe valve 1-FCV-110A	a. Fails open.	a. Boron Concentration Control - reactor makeup control - boration and auto makeup	a. Failure prevents the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the RCS coolant during normal plant operation, load follow and when bringing the reactor to a hot standby condition. Boration to bring the reactor to a hot standby condition is possible, however, flow rate of solution from BA tanks can not be automatically controlled.	a. Valve position indication (open to closed position change) at CB; and boric acid flow recording (FR-110) and flow deviation alarm at CB.	1. Same remark as that stated for item #4 in regards to valve design.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

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Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
	b. Fails closed.	b. Boron Concentration Control - reactor makeup control - boration, and auto makeup	b. Failure blocks fluid flow of boric acid solution from BA tanks during normal plant operation, load follow, and when bringing the reactor to a hot standby condition. Boration (at BA tank boron concentration level) of RCS coolant to bring the reactor to hot standby condition is possible by opening of alternate BA tank isolation valve (1-8104) at CB.	b. Valve position indication (closed to open position change) at CB; and boric acid flow recording (FR-110) and flow deviation alarm at CB.	
49. Air diaphragm operated globe valve 1-FCV-111A	a. Fails closed.	a. Boron Concentration Control - reactor makeup control - dilute, alternate dilute and auto makeup	a. Failure blocks fluid flow of water from reactor makeup control system during normal plant operation and load follow. No effect on system operation when bringing the reactor to a hot standby condition.	a. Valve position indication (closed to open position change) at CB; VCT level indications (LI-112A and LI-183) and low level alarms (low and low-low) at CB; and makeup water flow recording (FR-110) and flow deviation alarm at CB.	1. Same remark as that stated from item #7 in regards to valve design.
	b. Fails open.	b. Boron Concentration Control - reactor makeup control - dilute, alternate dilute and auto makeup.	b. Failure prevents the addition of a pre-selected quantity of water makeup at a pre-selected flow rate to the RCS coolant during normal plant operation and load follow. No effect on system operation when bringing the reactor to a hot standby condition.	b. Valve position indication (open to closed position change) at CB and makeup water flow recording (FR-110) and flow deviation alarm at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 32 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
50. Motor operated globe valve 1-8104	a. Fails closed.	a. Boron Concentration Control - reactor makeup control - boration and auto makeup	a. Failure reduces redundancy of flow paths for supplying boric acid solution from BA tanks to RCS via charging pumps. No effect on CVCS operation during normal plant operation, load follow, or hot standby operation. Normal flow path via RMCS remains available for boration of RCS coolant.	a. Valve position indication (closed to open position change) at CB and flow indication (FI-183A0) at CB.	1. Valve is at a closed position during normal plant RMCS operation. 2. If both flow paths from BA tanks are blocked due to failure of isolation valves (1-FCV-110A and 1-8104), borated water from RWST is available by opening isolation valve 1-LCV-112D or 1-LCV-112E.
	b. Fails open.	b. Boron Concentration Control - reactor makeup control - boration and auto makeup	b. Failure prevents the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the RCS coolant during normal plant operation, load follow and when bringing the reactor to a hot standby condition. Boration to bring the reactor to a hot standby condition is possible, however, flow rate of solution from BA tanks can not be automatically controlled.	b. Valve position indication (open to closed position change) at CB and flow indication (FI-183A) at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 33 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
51. Boric acid transfer pump #1 APBA-1 (BA transfer pump #2 analogous)	a. Fails to deliver working fluid.	a. Boron Concentration Control - reactor makeup control - boration and auto makeup.	a. No effect on CVCS system operation during normal plant operation, load follow or bringing reactor to hot standby condition. Redundant BA transfer pump #2 provides necessary delivery of working fluid for CVCS system operation.	a. Pump motor start relay position indication (open) at CB and local pump discharge pressure indication (PI-113).	
52. Solenoid operated valve 1-8210A (1-8210B analogous)	Fails open		Failure has no effect on PDP operation during normal plant operation and safe shutdown. However, under accident conditions requiring PDP suction stabilizer isolation, the failure reduces the redundancy of providing isolation of the gas makeup supply to the PDP suction stabilizer. Alternate isolation valve (1-8210B) provides backup isolation capability.	Valve position indication (open to closed position change). In addition, close position group monitoring light at CB.	The valve is designed to fail “closed” and is electrically wired so that the electrical solenoid is energized to open the valve. The solenoid is de-energized to close the valve upon generation of an ESF “S” signal.
53. Solenoid operated valve 1-802A (1-8202B analogous)	Fails open		Failure has no effect on PDP operation during normal plant operation and safe shutdown. However, under accident conditions requiring PDP suction stabilizer isolation, the failure reduces the redundancy of providing isolation of the gas vent line from the PDP suction stabilizer to the VCT. Alternate isolation (1-8202B) provides backup isolation capability.	Same methods of detection as those stated for item #52, failure mode “Fails Open”.	Same remark as stated for item #52 in regard to valve design.

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 34 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks
54. Air operated ball valve HV-8220 (HV-8221 analogous)	Fails open	a. Charging Pump suction high point vent.	Failure has no effect on CVCS operation during normal plant operation, load follow, and bringing reactor to a hot standby condition. However, under accident conditions requiring isolation of VCT, failure reduces the redundancy of providing isolation for vent line to the VCT.	Valve position indication (open to closed position change) at CB.	1. During normal plant operation and load follow, valve is at the full open position and the solenoid operator is de-energized to close the valve upon the generation of a VCT low-low water level signal or upon the generation of a safety injection “S” signal, providing isolation valve 1-LCV-112D (1-LCV-112E analogous) is at full open position.
	Fails closed	b. Charging Pump suction high point vent.	Failure blocks high point vent to VCT during normal plant operation, load follow, and when bringing the reactor to a hot standby condition.	Valve position indication (closed to open position change) at CB; group monitoring light (valve closed) at CB.	2. Valve solenoid operator is also de-energized to close the valve upon the generation of a source range neutron flux doubling signal, providing isolation valve LCV-112D (1-HV-8221 analogous) is at full open position.
55. Motor operated gate valve HV-8402A.	Fails closed.	Charging and Volume Control-charging flow.	Failure inhibits use of normal charging line to RCS for boration, dilution, and coolant makeup operations. Seal water injection path remains available for boration of RCS to a hot standby concentration level and makeup of coolant during operations to bring the reactor to hot standby condition.	Valve position indication (closed to open position change) and group monitoring light (valve closed) at CB.	

TABLE 9.3-9
FAILURE MODE AND EFFECTS ANALYSIS CHEMICAL AND VOLUME CONTROL SYSTEM ACTIVE COMPONENTS – NORMAL PLANT OPERATION AND LOAD FOLLOW

(Sheet 35 of 35)

Component	Failure Mode	CVCS Operation Function	Effect on System Operation and Shutdown[a]	Failure Detection Method[b]	Remarks

Note:

Unit 1 and Unit 2 Tag Numbers are generally the same except for the prefix or as otherwise noted

List of Acronyms and abbreviations

BRS	-	Boron Recycle System
BTR‘	-	Boron Thermal Regeneration
BTRS	-	Boron Thermal Regeneration System
CB	-	Control Board
CVCS	-	Chemical and Volume Control System
Demin.	-	Demineralizer
PRZ	-	Pressurizer
RC	-	Reactor Coolant
RCS	-	Reactor Coolant System
RHRS	-	Residual Heat Removal System
RWST	-	Refueling Water Storage Tank
RMCS	-	Reactor Makeup Control System
VCT	-	Volume Control Tank

a) See list at end of table for definition of acronyms and abbreviations used.

b) As part of plant operation, periodic tests, surveillance inspections and instrument calibrations are made to monitor equipment and performance. Failures may be detected during such monitoring of equipment in addition to detection methods noted.

9.4 AIR CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEMS

9.4.1 CONTROL ROOM AREA VENTILATION SYSTEM

9.4.1.1 Design Bases

The Control Room HVAC and filtration systems are designed to maintain suitable and safe ambient conditions for operating personnel and equipment during all modes of operation including post-DBA conditions, in the following areas of the Control Building.

Areas on floor elevation 830 ft 0 in.:

East Control Room

West Control Room

Console and Control Room Unit 1

Console and Control Room Unit 2

Instrument Room Unit 1

Instrument Room Unit 2

Computer Room Unit 1

Computer Room Unit 2

Shift Manager’s Office

Production Supervisor’s Office

Corridor

Men’s Toilet

Woman’s Toilet

Kitchen and Janitor Closet

Charts and Supplies Storage Room

Areas on floor elevation 840 ft 6 in.:

Technical Support Center (Office and Corridor)

Offices (2)

Electrical Equipment Corridor

Areas on floor elevation 854 ft 4 in.:

Control Room Air Conditioning System mechanical equipment rooms, trains A and B.

The Control Room, located on elevation 830 ft 0 in, is maintained at 75°F (±5°F) and 35-50 percent relative humidity. The Control Room HVAC and filtration equipment rooms are maintained between 40°F and 104°F. Miscellaneous areas on elevations 830 ft and 840 ft are also maintained between 40°F and 104°F. Other system design parameters are presented in Tables 9.4-1 and 9.4-2.

Design Conditions-Indoors in Table 9.4-2 were based on personnel comfort or equipment design temperature limitation. Heating, ventilating and air-conditioning equipment were sized to provide sufficient heating and cooling during maximum and minimum outdoor conditions as indicated in Table 9.4-1. Temperatures and wind speed identified in Table 9.4-1 were selected on the bases of the meteorological study presented in Section 2.3. More specifically the maximum temperatures are based on Table 2.3-15. The extreme temperatures in Table 9.4-1 were not reached during the period of record indicated in Table 2.3-15. The duration of such extreme temperatures is considered short (one to two hours).

As described in the following paragraphs, the system is provided with sufficient redundancy in equipment and power supplies to enable the system to sustain a single failure of an active component without loss of function.

1. The system is equipped with four modular air-conditioning units. Each unit is rated at 50 percent of the Control Room HVAC and filtration systems capacity. Each pair of air-conditioning units is powered from an independent Class 1E bus and is physically separated by a concrete wall in the Control Room HVAC and filtration mechanical equipment room.

2. The Control Room makeup air supply fans, Control Room exhaust fans, Control Room complex kitchen and toilet exhaust fans, emergency pressurization air supply filtration units and fans, and emergency filtration units and fans are 100-percent redundant.

3. The system is equipped with ten non-safety related local air-conditioning units that provide supplementary cooling to areas which do not contain safety related equipment and are not needed for continued occupancy.

4. The redundant fans and filtration units are powered from independent Class 1E buses. (See Section 8.3.)

5. Redundant outside air intake dampers CPX-VADPOU-14 and CPX-VADPOU-15 (Figure 9.4-1) are provided to supply air during normal operation and to allow air for pressurization during emergency conditions.

6. All dampers are set to fail in the safe position or are provided with separate, bottled air supplies for emergency operation. (See Section 9.3.1)

7. All exhaust fans are equipped with isolation dampers CPX-VADPMU- 05,06 and CPX-VADPOU-27,28 (Figure 9.4-1).

8. Dampers CPX-VADPOU-41 and CPX-VADPOU-42 are provided for the isolation of the emergency filtration units.

9. The condensers for each pair of air-conditioning units have an independent cooling water source.

10. All control valves and dampers are equipped with manual operators at accessible locations to facilitate their operation in the event of power or instrument air failures, or both.

11. Redundant inlet bubble tight dampers are provided at each control room inlet to minimize unfiltered inleakage when the system is in the emergency mode of operation and isolation during toxic gas release or fires.

System components and ductwork are of seismic Category I and ANS Safety Class 3 design to ensure system availability for safe shutdown of the reactor following DBAs.

Failure modes for isolation valves and dampers are set so that their failure does not render the system inoperable.

Radiation protection (see Sections 11.5 and 12.3.3) is provided to permit access and occupancy of the Control Room during normal plant operation and following DBAs without personnel receiving radiation exposures in excess of five rem whole body, or its equivalent, to any part of the body for the duration of the accident, which is in accordance with GDC 19 of 10 CFR Part 50, Appendix A.

Surveillance of activity levels at the fresh air intake ducts to the Control Room is provided by seismically qualified process radiation monitors which are shown on Figure 9.4-1. (See Section 11.5 for design criteria.) The activity level at each intake duct is monitored and indicated in the Control Room; high levels are annunciated on the radiation monitoring system display/operator cabinet.

Continuous surveillance of radiation dose levels in the Control Room is also provided by redundant area radiation monitors which are shown on Figure 9.4-1. (See Section 12.3.4 for design criteria. Radiation dose level is continuously monitored, and indicated in the Control Room and annunciated on the radiation monitoring CRT display console.

9.4.1.2 System Description

The Control Room design shown in Section 1.2, Figure 1.2-33, provides for a single Control Room serving both units. The Control Room HVAC system is designed to serve this area as well as the computer rooms and offices located at the same elevation and at elevation 840 ft 6 in. The Control Room HVAC system is designed to remove all heat generated by the equipment, computers, lighting, personnel, and so forth and to provide a safe operating condition for personnel.

Two modular air-conditioning units provide the cool air required, with two units of the same size on standby to address the single failure criterion. Each modular air-conditioning unit contains a roughing filter, cooling coil, heating element, humidity control equipment, fan, and associated instrumentation controls.

The cooling coils are the direct-expansion, refrigerant type and suitable for refrigerant R-12. The coils of each modular air-conditioning unit are connected to separate and independent compressors and water-cooled condensers.

During normal operation, the air is recirculated through the air-conditioning unit with about six percent outdoor air (3000 ft^3/min fresh air) added to provide for oxygen depletion, odor control, pressure boundary leakage, and provide makeup for the kitchen and toilet and Control Room Exhaust Fans. Makeup air is supplied by one of two redundant fans.

Two redundant emergency filtration units are provided for use following DBAs. Each unit comprises a roughing filter, two high-efficiency particulate absorption (HEPA) filters, and iodine adsorbers and booster fans.

Exhaust air from the toilets, and kitchen areas is discharged directly to the atmosphere through one of the two 100-percent-capacity toilet exhaust fans.

The Control Room is pressurized with respect to the environment to prevent infiltration of unfiltered and unmonitored air and to account for leakages, as specified in Section 6.4. The overpressure is considered sufficient to prevent infiltration because the Control Room building structure is completely airtight, with few penetrations, and designed to withstand the tornado-generated missiles described in Section 3.3.2. This overpressure is maintained by modulating exhaust damper CPX-VADPMU-05 or CPX-VADPMU-06 during normal operation and emergency ventilation. Damper CPX-VADPOU-22 or CPX- VADPOU-23 can be used to reduce the control room pressure down to .125 overpressure for easier access to the control room during emergency recirculation. (See Section 6.4.)

Overpressure during the emergency recirculation mode is accomplished by introducing up to 800 ft^3/min of filtered air into the Control Room. This air is supplied through a 100-percent-redundant air-cleanup unit consisting of a fan, a mist eliminator, a heater, a roughing filter, two HEPA filters, and an iodine adsorber.

Section 6.4.2 and Table 6.4-4 describe the potential leak paths and leakage characteristics.

The orientation of the Control Room doors (with the exception of the missile resistant door) to swing inward is necessary to reduce the exfiltration rate from the Control Room and to prevent a possible depressurization of the Control Room caused by a door being left ajar. Provision of a positive pressure against an inward-opening, gasketed door results in a significant decrease in leakage over an outward-opening door. Consequently, a reduced emergency pressurization flow rate is required during possible post-LOCA operation. The lower outside air intake rate results in a Control Room operator dose well below the allowable limits of GDC 19, 10 CFR Part 50 Appendix A. For an analysis of the dose received by the Control Room occupants in the unlikely event of a LOCA, see Section 15.6.5.4

Operation of the Control Room HVAC system comprises the following modes:

1. Normal operation

2. Emergency recirculation

3. Emergency ventilation

4. Isolation

The four modes of operation are shown on Figure 9.4-1.

During normal operation, one makeup air supply and one main exhaust fan are operating, and their associated dampers are open. The emergency filtration system is isolated, and its bypass damper is open. One kitchen and toilet exhaust fan also operates.

The Control Room HVAC system automatically switches to emergency recirculation when any one of the conditions outlined in Subsection 9.4.1.3 are detected. The procedure is as follows:

1. The makeup air supply fan and all exhaust fans stop.

2. All exhaust dampers close.

3. The isolation and bypass dampers of the emergency filtration units are positioned such that approximately 16 percent (8000 ft^3/min) of the recirculated air flow is directed through each of the filters and booster fans.

4. Both emergency filtration unit fans start.

5. Both emergency pressurization supply unit fans start. The dampers are positioned such that the air is directed through the pressurization air cleanup units to the emergency filtration unit.

In cases 4 and 5 above, the operator, within one (1) hour, stops one pressurization and one filtration unit fan. Redundant inlet isolation dampers are closed to minimize filtered inleakage through the stopped train.

The operation of the emergency ventilation mode is required to replenish the oxygen content in the Control Room atmosphere during the emergency recirculation mode. For brief intervals the makeup fan is operated, supplying filtered outside air (3800 ft^3/min) to the Control Room complex. The main exhaust fan and the kitchen and toilet exhaust fan are also operated simultaneously.

Subsequent to the automatic initiation of the emergency recirculation mode, the operator can regain manual control over the system from the Control Room ventilation panels.

The operator may take manual action to transfer the control room ventilation system from the normal mode to the isolation mode following control room annunciation of smoke at either of the two outside air intakes (see Section 6.4). The return to the normal operation mode also requires manual action

Each Control Room air-conditioning unit is an independent, seismic Category I package unit with a component cooling water cooled condenser. Electric reheat coils are provided for the the production supervisors office, and the main console areas and are powered by the non-safety-related electrical bus.

9.4.1.3 Safety Evaluation

The Control Room atmosphere cleanup units (integrated into the HVAC system design) conform to the criteria established in NRC Regulatory Guide 1.52, Reference [10], except as shown in Appendix 1A(B) and Section 6.5, Table 6.5-1.

The Control Room HVAC system is provided with instrumentation and controls which continuously monitor system performance. In addition, area radiation and fire detection monitors are provided to ensure safe operating conditions for equipment and personnel during all modes of operation.

An ionization detector is located in each fresh air intake to detect smoke in the incoming air. If smoke is detected, the respective inlet and outlet Control Room air duct dampers may be closed and air recirculated in the Control Room. Switching to the normal or emergency ventilation modes of operation is accomplished manually by the operator by means of a Control Room vertical panel-mounted selector switch.

Any contaminants that have entered the Control Room prior to full closure of the dampers are removed by the emergency filtration unit. Removal of heavy concentrations of contaminants caused by a fire in the Control Room is accomplished by portable smoke ejectors.

Audible and visual alarms are provided in the Control Room to alert the operator in the event of system malfunction or unsafe conditions.

The Control Room air-conditioning system is designed to automatically switch to the emergency recirculation mode of operation described in Subsection 9.4.1.2 should the offsite power fail (for operator convenience only). The system also automatically switches to this mode of operation upon receiving a Control Room ventilation high-radiation signal or a safety injection signal from either Unit 1 or Unit 2. The radiation signal originates from radiation monitors which sample the Control Room intake air vents. Radiation detectors of this type are discussed in Section 11.5.

The iodine absorbers are manufactured from impregnated activated carbon and are used for radioiodine compound removal.

The iodine adsorbers construction and efficiencies comply with NRC Regulatory Guide 1.52 (See Appendix 1A(B)). Additional data concerning iodine adsorbers is presented in Table 9.4-6.

Anticipated efficiencies of filters are in accordance with Table 9.4-4.

An analysis of postaccident dose levels in the Control Room is presented in Section 15.6.5.4.

Sufficient redundancy in equipment and power supplies enables the system to sustain a single failure without total loss of function. The emergency pressurization and recirculation filter trains, plus supply and exhaust fans and the associated dampers, are completely redundant. Four parallel air-conditioning units are used to provide a 100 percent (two air conditioners) standby feature.

Redundant, bubble tight inlet isolation dampers are provided to minimize filtered inleakage via a non-running train.

The probability of an electrical fire in the Control Room is low because of the low-voltage and flame-retardant cables used. Fire dampers automatically isolate the affected areas to prevent spreading of the fire and shut off the oxygen supply. The charcoal adsorbers are also provided with a fire protection water spray system. Actuation of the system also activates an alarm in the Control Room. A failure mode and effects analysis is presented in Table 9.4-8.

Alarms in the Control Room alert the operator to any system malfunction so that he can manually actuate the necessary standby units. The maximum operational temperature limit for Control Room instrumentation is 120°F. This temperature is not reached during the short periods of system malfunction prior to the actuation of the standby units.

The Control Room HVAC equipment is located in a seismic Category I structure above the Control Room at elevation 854 feet 4 inches, as shown in Section 1.2, Figure 1.2-6. The Control Room HVAC system is seismic Category I and ANS Safety Class 3. Outside air enters through intake louvers located in the north and south walls of this structure. Concrete walls and slabs, behind the primary wall, are located so as to prevent postulated missiles from entering further into the building. Refer to Figure 6.4-3. Air flow into the building is provided through an opening in one of the tornado missile enclosure walls; the opening is so located that no postulated missile can pass through into the building, thereby protecting the outside air intake ducts and mechanical equipment from tornado-related missiles described in Section 3.3.2. Any damage to an intake louver will not impair the functioning of the air intake.

All safety-related Control Room HVAC equipment required for operation during a loss of offsite power or following a LOCA is powered from the redundant Class 1E buses.

9.4.1.4 Inspection and Testing Requirements

Shop inspection and testing are performed for all equipment, including heating and cooling coils and controls.

The system is initially tested and adjusted for proper flow paths, flow capacities, heating and cooling capacities, mechanical operability, and filter efficiency.

Fans are rated and tested in accordance with the standards of the Air Moving and Conditioning Association (AMCA).

The HEPA filters and iodine adsorbers are tested periodically during plant operation, in accordance with NRC Regulatory Guide 1.52, (see Appendix 1A(B)). Filter units are arranged to facilitate cell replacement.

Heating and cooling coils are tested in accordance with manufacturer's standards. Ductwork and filter are tested in accordance with ANSI N510 and industry standards.

Redundant standby equipment is operated on a cyclic basis to ensure the availability of the equipment.

The inlet bubble tight dampers are shop tested in accordance with ASME CODE AG-1 and then tested periodically as part of the ventilation system, to ensure minimum filtered inleakage into the control room atmosphere when expected to perform their isolation function.

9.4.2 SPENT FUEL POOL AREA VENTILATION SYSTEM

The spent fuel pool area ventilation system is incorporated in the Fuel Building ventilation system and the Primary Plant Ventilation System.

9.4.2.1 Design Bases

The Fuel Building ventilation system is designed to maintain suitable ambient conditions for personnel and equipment during normal plant operations and scheduled shutdowns. Ambient temperature throughout the building is normally maintained as shown in Table 9.4-2. During emergency conditions (LOCA with a loss-of-offsite power) the ambient temperature in the spent fuel pool heat exchanger and pump rooms shall be maintained below 122°F, though the temperature may rise to 129°F for a short duration. System design parameters are shown in Table 9.4-1.

In addition, a slight negative pressure is maintained during normal operation and during a fuel handling accident to prevent the outflow of unfiltered, contaminated air to the environment. Operating the primary plant ventilation exhaust filter trains minimizes the release of radioactive particulate effluents and radioiodine to the environment.

The dissipation of heat from the spent fuel pool cooling pumps during loss of offsite power is accomplished by using emergency fan coil units, which maintain the safety-related pump rooms below the maximum ambient temperature allowed by the equipment design. The emergency fan coil units are supplied with chilled water from the safety-related chilled water system. (See Figure 9.4-12.) The HVAC equipment is designed to operate in and to maintain the required ambient conditions. The supply and exhaust ventilation units are a part of the modular arrangement for the primary plant ventilation system as shown in Figure 9.4-9 and are described in Subsection 9.4.3.

Wherever possible in the Fuel Building, airflow is directed from areas of lower potential radioactivity to areas of higher potential radioactivity.

Tritium concentration buildup and dispersion throughout the Fuel Building is reduced by a local network of exhaust ducts at the spent fuel pool, as shown schematically on Figure 9.4-2. The spent fuel pool area ventilation system is designed to reduce operator dose as a result of the evaporation of tritiated water. Fresh air is supplied over the spent fuel pool and exhausted by exhaust ducts located near the pool surface. This air is passed through a demister and booster fan before entering the primary plant ventilation exhaust system.

Sufficient air changes are provided to maintain the concentration of airborne radioisotopes throughout the area during fuel handling operation below the concentration levels specified in Appendix B to 10 CFR Part 20[a]. (See Section 12.2.2.)

The modular exhaust filtration unit arrangement provides sufficient redundancy in equipment and power supplies for the system to sustain a single active component failure without loss of function. Each emergency fan coil unit is interconnected with the same Class 1E bus as the

a. Design assessment made with the provisions of 10 CFR 20.1-20.601.

equipment it serves. Instrumentation and controls which incorporate visual as well as audible alarms are provided enabling the operator to continuously monitor system performance and manually switch to the standby fan coil unit when required.

The exhaust air is passed through iodine adsorber beds and discharged through the plant ventilation discharge vent, where it is monitored for activity. An in-line detector monitors ventilation air at the outlet of the Fuel Building exhaust header. Data concerning iodine adsorbers are presented in Tables 9.4-4 and 9.4-6. Data concerning prefilters and HEPA filters are presented in Table 9.4-4. The exhaust air ductwork for the fuel pool area is embedded in the walls of the fuel pool, with the bottom of the exhaust registers at approximately six in. above the water surface. Air delivered by the primary plant supply units (supply air outlets located approximately 30 ft) above the pool surface is drawn downward over the surface through the exhaust registers. This arrangement inundates the occupied areas with contaminant-free air, thus minimizing personnel exposure to contaminated air.

Mist eliminators (demisters) are provided in the spent fuel pool exhaust to remove moisture in the exhaust air that exists as a result of evaporation of high-temperature fuel pool water. The entrained water is routed by a drain system to a Fuel Building sump. In addition, care is taken not to impair the visibility for underwater observation by using the following:

1. Maintaining a low sweep velocity over the spent fuel pool surface to prevent ripple propagation.

2. Supplying sufficient air at a temperature calculated to avoid saturation of the air at the spent fuel pool surface.

9.4.2.2 System Description

The Fuel Building ventilation system is shown schematically on Figure 9.4-2. The primary plant air supply system delivers filtered and tempered outside air to each floor of the Fuel Building through a duct distribution system. Each air supply unit consists of a roughing filter, heating and cooling coils, and a 100-percent-capacity fan.

The arrangement of filters and HVAC equipment is in accordance with the requirements of NRC Regulatory Guide 8.8, Revision 2, to achieve ALARA radiation levels. (See Figure 9.4-9.) Air supplied to the spent fuel pool area is heated via an electric heater to prevent fogging at the pool surface. The cooling water supply to the PPV supply unit cooling coils is maintained by the plant ventilation chilled water system only during normal operation. Chilled water from the plant ventilation chilled water system is shown in Figure 9.4-11.

The fuel handling area ventilation system is designed to maintain a slight negative pressure with respect to the environs during normal operation, LOCA or loss of offsite power.

The exhaust system ductwork branches to all areas within the fuel handling area. Exhaust air is discharged to the atmosphere through the plant ventilation discharge vent after passing through the primary plant modular exhaust filtration units. Each modular non-ESF exhaust filtration unit consists of a roughing and two HEPA filters, an iodine adsorber, and a fan. Mist eliminators are provided in the pool exhaust to remove entrained moisture particles from the exhaust air stream. Trapped moisture from these mist eliminators is returned to the spent fuel pool sump via a system of drains.

The Fuel Building exhaust is routed through the Primary Plant Ventilation exhaust system. (See Figures 9.4-2 and 9.4-9).

Emergency fan coil units are provided for the spent fuel pool pump rooms. The fan coil units maintain suitable operating temperatures in the pump rooms during all modes of operation. The cooling medium for these fan coil units is the nuclear safety chilled water system.

A room mounted air handling unit is provided for Storage Room 250A. The air handling unit maintains adequate cooling for non-safety related equipment which is located within the room. The cooling medium for the air handling unit is chilled water which is provided to the cooling coil by a yard mounted air cooled chiller.

9.4.2.3 Safety Evaluation

The fuel handling area ventilation system conforms to the regulatory position established in NRC Regulatory Guide 1.13.

The system complies with the environmental and components design criteria, the qualification testing provisions, and the intent of the system design criteria promulgated in NRC Regulatory Guide 1.140. Even if the system fission-product removal features failed to function properly, the exclusion area boundary doses resulting from a postulated fuel handling accident would still be well below the exposure guidelines of 10 CFR Part 100. (See Section 15.7.4.3 for an analysis of a postulated fuel handling accident.)

During an abnormal condition, as the spent fuel pool water temperature increases from the normal operating temperature the evaporation rate from the pool increases. See Table 9.4-7. For this purpose, mist eliminators are provided in the exhaust from the spent fuel pool areas. The charcoal and the HEPA filters are protected from free water particles by these demisters and from high humidity by the mixing of the Fuel Building exhaust air with the Auxiliary and Safeguards buildings exhaust air.

The Fuel Building is completely separate from the Containment, and its function is totally independent of the reactor operation; it is not required for safe plant shutdown in the event of a severe natural phenomenon or LOCA. Therefore, during a LOCA the normal Fuel Building (i.e., primary plant) ventilation system is shut down, but it remains in operation after a fuel handling accident.

Alarms are provided to alert operators of equipment malfunction and to facilitate the startup of standby units. Radiation levels are continuously monitored. (See Sections 11.5 and 12.3.4).

During fuel handling operations, the system is sized to maintain the concentrations of airborne radioisotopes below the concentration levels specified in Appendix B to 10 CFR Part 20[a]. (See Section 12.2.2.) For a discussion of a fuel handling accident, see Section 15.7.4.3.

The primary plant ventilation system is described in Section 9.4.3.

The Fuel Building air exhaust ductwork is ANS Safety Class 3, seismic Category I, and NNS, seismic Category II. The emergency fan coil units, which are located in the safety related pump rooms, are seismic Category I and ANS Safety Class 3. The ventilation failure modes are

presented in Table 9.4-9. The spent fuel pool exhaust fans are non- nuclear safety related and non-seismic. Operation of these fans is not required to limit Fuel Building exfiltration during normal operation. The Operator verifies the exhaust fans automatically trip following a safety injection signal and manually trips the fans as required.

9.4.2.4 Inspection and Testing Requirements

Shop inspection and testing are performed for all equipment, including heating and cooling coils and controls.

The system is initially tested and adjusted for proper flow paths, flow capacities, heating and cooling capacities, mechanical operability, and filter efficiency.

Fans are rated and tested in accordance with the standards of the Air Moving and Conditioning Association (AMCA).

Heating and cooling coils are tested in accordance with manufacturer's standards. Ductwork and filter are tested in accordance with ANSI N510 and industry standards.

Redundant standby equipment is operated on a cyclic basis to ensure the availability of the equipment.

9.4.3 AUXILIARY BUILDING AND RADWASTE AREA VENTILATION SYSTEM

9.4.3.1 Design Bases

Auxiliary Building (controlled access area) and radwaste area ventilation system is designed to maintain suitable and safe ambient conditions for operating equipment and personnel during normal plant operation.

During normal plant operation and during and after a LOCA, a slightly negative pressure is maintained. The dissipation of heat during abnormal conditions is accomplished by using emergency fans or fan coil units which maintain the safety-related equipment rooms that need cooling below the maximum permissible ambient temperature. The HVAC equipment is designed to operate in and to maintain the required ambient conditions. The supply and exhaust ventilation filtration units are a part of the modular arrangement for the primary plant (controlled access area) ventilation system as shown on Figures 9.4-9 and 9.4-2.

Ambient temperatures throughout the building are normally maintained as shown in Table 9.4-2. During loss of offsite power and LOCA, safety related equipment rooms equipped with emergency fans or fan coil units are kept below 122°F though the temperature may exceed 122°F for a short duration. However, the maximum indoor temperature during loss of offsite power and LOCA is expected to be kept below 131°F. Other system design conditions are presented in Tables 9.4-1 and 9.4-2.

The ductwork layout is arranged so that in areas where airborne radioactivity may be present, airflow is directed from areas of lower potential radioactivity toward areas of higher potential radioactivity. The waste gas evaporator and waste gas compressor areas are provided with direct exhaust duct connections to the exhaust filter units, while the supply air originates from the surrounding areas of lesser possible radioactivity.

The gas decay tanks are located in separate, closed compartments; flexible hose duct connections are provided for attachment to a manifold in the exhaust systems when it is necessary to ventilate these compartments. The compartments are not ventilated unless personnel access is required. The compartments can then be loaded manually and individually onto the ventilation system.

The Recycle Holdup Tanks (see Section 9.3.4) are also located in separate compartments. Should the flexible diaphragms in the tanks rupture, the hydrogen gas released would be contained in the rooms until the ventilation system has removed it or diffusion has reduced it below the lower explosive limit. All penetrations are sealed, except between compartments, and electrical equipment is explosion proof above the elevation 830 feet in the compartments.

The airflow quantities are adequate to ensure that the concentration of radioisotopes in all areas, including the radwaste area, is below the concentration levels specified in Appendix B to 10 CFR Part 20[a].

Emergency fan coil units are provided for cooling the centrifugal charging pump and component cooling water pump rooms. The units are sized to remove the heat dissipated by the pumps and are powered from the same electrical safety-related buses as the components they serve. The cooling units are supplied with chilled water from the plant safety chilled water system (as shown on Figure 9.4-12) during normal operation, during a LOCA, and after a loss of offsite power.

Each emergency fan coil unit is interconnected so that it starts with the equipment it serves. The system is provided with sufficient instrumentation to enable the operator to continuously monitor the system performance and manually switch to the standby units when required.

Differential pressure measurement equipment is provided to ensure that a slightly negative pressure with respect to the environs is maintained during normal operation and during and after a LOCA in the building, preventing the outward leakage of unfiltered contaminated air to the environment.

Exhaust air is passed through HEPA filters and iodine adsorber beds contained in the modular exhaust filtration units of the Primary Plant Ventilation System. Its activity is monitored prior to discharge to the atmosphere. The estimated annual radioactivity released from this source is discussed in Section 11.3.

9.4.3.2 System Description

The Auxiliary Building (controlled access area) and radwaste area ventilation system is shown schematically on Figure 9.4-2. The primary plant air supply system delivers filtered and tempered outside air to each floor of the Auxiliary Building through a duct distribution system. A minimum of three air supply units of the Primary Plant Ventilation System are normally useed for ventilation of the controlled access areas of the Auxiliary, Fuel and Safeguards Buildings.

Each air supply unit consists of a roughing filter, electric heating and cooling coil, and a fan. The arrangement of filters and HVAC equipment is in accordance with the requirements of NRC Regulatory Guide 8.8, Revision 2, to achieve ALARA radiation levels. Chilled water is supplied to the supply air handling units by the plant ventilation chilled water system, as shown by Figure 9.4-11. This system provides chilled water only during normal operation. The Auxiliary

Building ventilation system is designed to maintain a slightly negative pressure with respect to the environs during normal operation and during and after a LOCA. The exhaust system ductwork branches to all areas within the Auxiliary Building. Wherever possible, airflow is directed from areas of lower potential radioactivity to areas of higher potential radioactivity. Unit heaters are provided in specific areas for operator comfort. Exhaust air is discharged to the atmosphere through the plant ventilation discharge duct after filtration through the modular exhaust filtration units. The Auxiliary Building ventilation duct is monitored by an in-line radiation monitor at the outlet of the ventilation duct header. Each modular non-ESF exhaust filtration unit consists of a roughing and two HEPA filters, an iodine adsorber, and a fan.

Fan coil units are provided for the charging and Component Cooling Water pump rooms in the Auxiliary Building. The fan coil units maintain pump room temperature at a suitable level during all modes of pump operation. All emergency fan coil units are seismic Category I and ANS Safety Class 3 and use chilled water supplied by the nuclear safety chilled water system as a cooling medium. The positive displacement charging pump fan coil unit is non-safety-related and is supplied cooling water from the plant ventilation chilled water system.

The Fuel Building, Auxiliary Building, Safeguards buildings, and Containment purge supply and exhaust units are all part of a modular ventilation system. This system consists of twelve identical non-ESF exhaust ventilation units, four ESF exhaust units, and eight air supply units. Each of the non-ESF exhaust ventilation units consists of a roughing filter, HEPA filter, iodine adsorber, HEPA filter, and fan. Each of the eight air supply units consists of a roll filter, heating and cooling coils, and a fan. The total capacity provided is more than required for normal operation; thus adequate redundancy is provided to prevent a single supply or exhaust fan failure from affecting system operation. In addition, supply fans CPX-VAFNAV-17 through CPX-VAFNAV-22 are interlocked with exhaust fans CPX-VAFNCB-09 through CPX-VAFNCB-20 to prevent a possible pressurization of the plant in the event of insufficient exhaust fan capacity. Supply fans CPX-VAFNAV-23 and 24 are not interlocked with any of the exhaust fans.

During a Loss of Offsite Power, the non-ESF exhaust fans trip automatically causing the interlocked supply fans to also trip. The operator manually trips any remaining operating supply fans and energizes one (or two) Non-ESF exhaust fans to ensure monitoring of effluents (see Section 11.5.2.6.3).

During a LOCA, in either Unit 1 or Unit 2, all supply and exhaust fans trip automatically on the safety injection signal while the ESF exhaust fans start automatically. This trip is redundant and meets the single active failure criteria. This will ensure the primary plant ventilation system maintains a slight negative pressure in the negative pressure envelope during all modes of operation.

The supply and exhaust requirements for each building is shown on Figure 9.4-9. Interconnections between the primary plant modular ventilation system and all other buildings are shown on Figure 9.4-13.

Chilled water for the cooling coils is provided by the plant ventilation chilled water system. (See Section 9.4E). Heating coils are electric. The exhaust system is seismic Category I up to the fan discharge. The air supply system is non-nuclear safety seismic Category II except for the fan discharge gravity dampers that shall be safety Class 3 and seismic Category I.

9.4.3.3 Safety Evaluation

The reliability and safety of the Auxiliary Building ventilation system is ensured by the following features:

1. Instrumentation and controls which incorporate audible and visual alarms in the Control Room facilitate continuous monitoring of performance and alert the operator in the event of system malfunction.

2. Standby supply units can be remotely actuated from the Control Room.

3. Deleted

4. The Auxiliary Building exhaust system is of ANS Safety Class 3 and seismic Category I, and NNS, seismic Category II design.

5. Failure modes for dampers are set so that they do not render the system inoperable.

6. Adequate cooling of safeguard equipment is ensured by the operation of the emergency fan coil units.

7. Exhaust air is passed through iodine adsorber filter beds prior to its discharge. It is also monitored by an in-line radiation monitor at the ventilation duct outlet.

8. The safety Class 3 and seismic Category I gravity dampers at the fan discharge of the primary plant ventilation system and the Auxiliary Building Ventilation Equipment Room and the redundant trips of the supply and exhaust fans as discussed in Section 9.4.3.2 will insure the negative pressure envelope boundary.

The Auxiliary Building and radwaste area ventilation exhaust is located in the primary plant ventilation equipment room at elevations 873 ft 6 in. and 886 ft 6 in. of the Auxiliary Building, as shown in Section 1.2, Figure 1.2-35. The Auxiliary Building is a seismic Category I structure, and the air inlet and duct leading to the Auxiliary Building ventilation system are designed to appropriate seismic requirements and to withstand the tornado loads and tornado-related missile conditions described in Section 3.3.2. The intake louvers are facing west, toward the Turbine Building.

9.4.3.4 Inspection and Testing Requirements

1. See Subsection 9.4.2.4.

2. The HEPA and iodine adsorbers of the non-ESF units are tested periodically during plant operation in accordance with NRC Reg. Guide 1.140 (See appendix 1A(B)). Filter units are arranged to facilitate filter replacement.

9.4.4 TURBINE BUILDING AREA VENTILATION SYSTEM

9.4.4.1 Design Bases

The Turbine Building is provided with outside air as required for ventilation and heat removal. (See Figure 9.4-3 and Tables 9.4-1 and 9.4-2.)

The Turbine Building ventilation system is designed to provide sufficient heat to ensure freeze protection of equipment during the winter months and to provide a sufficient airflow quantity in order to remove the heat generated by equipment located in the Turbine Building. The Turbine Building ventilation system is not required to operate following a DBA or loss-of-offsite-power condition.

9.4.4.2 System Description

Each Turbine Building ventilation system consists of vane axial supply fans, roof-mounted exhaust fans with gravity dampers, and propeller-type wall ventilators used for exhaust. The Turbine Building ventilation system also consists of centrifugal supply fans and its ductwork for ventilating the heater drain pump and condensate pump areas. Mushroom type caps are used on some of the roof exhausters which prevent a direct upblast of air. In addition, there are related ductwork, dampers, and intake louvers, which complete the ventilation system. The HVAC system is designed in accordance with the requirements of NRC Regulatory Guide 8.8, Revision 2, to achieve ALARA radiation levels.

The Turbine Building has exhaust roof ventilators installed on a removable steel plate over the Main Feed Pump area to provide additional exhaust capacity. The air is drawn from the louvers on the west end of the building and from areas below.

The supply fans and ductwork are used to supply outdoor air to the areas below elevation 830 ft. including the condensate pump area and Unit 2 heater drain pump area.

The ventilation air from the lower elevations stratifies through grating to the grade elevation above 803 ft 0 in. (the mezzanine level of the Turbine Building). Outdoor air enters the mezzanine level of the Turbine Building through multiple combination louver-dampers mounted on the west wall of the Turbine Building. The outdoor air which enters the mezzanine level and the air that rises from the lowest elevation of the Turbine Building is exhausted through wall exhausters and roof-mounted exhaust fans.

Locations of the exhaust vents are shown on Figures 1.2-3, 1.2-5, 1.2-24 and 1.2-29. Discharge temperature is dependent on ambient temperature conditions.

Electric unit heaters are provided as a source of auxiliary heat required during the normal modes of plant operation or during maintenance outages. The electric unit heaters are located around the perimeter of the mezzanine level.

The Turbine Building lube oil reservoir room is provided with a wall-mounted intake louver and exhaust fan to adequately ventilate this area. Ambient air from the Turbine Building is drawn through the intake louver and exhausted to the outside by a wall ventilator.

The turbine oil room at elevation 778 ft 0 in. is provided with a wall-mounted intake louver and an exhaust system including a wall-mounted fan and associated ductwork. Ambient air is drawn from the basement elevation of the Turbine Building through a wall-mounted intake louver by a wall-mounted exhaust fan. The exhaust fan is connected to an exhaust duct which discharges the room air to the atmosphere.

The thyristor voltage regulator room is provided with an air-cooled, wall-mounted, air-conditioning unit and backup A/C unit which recirculates and cools the ambient air of this solid state electronics equipment room. The Digital Control Room is provided with an air-cooled, air-conditioning unit and a back-up air-conditioning unit which recirculates and cools the ambient air of this solid state electronic equipment room. A wall-mounted intake louver and a wall-mounted exhaust fan are provided to ensure adequate ventilation of the machinery room. Air is drawn from the basement elevation of the Turbine Building through the intake louver by the wall-mounted exhaust fan. The exhaust fan discharges the ambient air back into the basement elevation of the Turbine Building.

All enclosed stairwells in the Turbine Building are pressurized by roof-mounted supply fans during normal operation to prevent smoke infiltration in the event of a fire.

9.4.4.3 Safety Evaluation

The reliability of the Turbine Building ventilation system is ensured by the following features:

1. The use of multiple roof exhaust fans with excess capacity and power supplies enables the system to sustain a single failure of any single fan without adversely affecting the overall ventilation of the Turbine Building.

2. Instrumentation and controls incorporating audible and visual annunciation facilitate continuous monitoring of the system performance and alert the operator in the event of a system malfunction.

3. There is no treatment of the Turbine Building exhaust air because the discharge from the condenser vacuum pumps is routed to the plant vent stack via the controlled access modular exhaust units and therefore these turbine building exhausts are not release points. (See Section 10.4.2.2.)

No radiation detectors monitor the Turbine Building exhaust per se, but the vacuum pump discharge is monitored. (See Section 10.4.2.3.) The Turbine Building ventilation system is non-seismic Category I and non-nuclear safety.

9.4.4.4 Inspection and Testing Requirements

1. Shop inspection and testing are performed for all equipment, including heating and cooling coils and controls.

2. The system is initially tested and adjusted for proper flow paths, flow capacities, heating and cooling capacities, mechanical operability, and filter efficiency.

3. Fans are rated and tested in accordance with the standards of the Air Moving and Conditioning Association (AMCA).

4. Ductwork is tested in accordance with industry standards.

9.4.5 ENGINEERED SAFETY FEATURES VENTILATION SYSTEM

The engineered safety features (ESF) ventilation system is incorporated as parts of the building ventilation systems (Building exhaust and emergency fan coil units).

9.4.5.1 Design Bases

The Safeguards Building ventilation system is designed to maintain suitable and safe ambient conditions for operating equipment and personnel during normal plant operation and maintains portions of the building under a slightly negative pressure with respect to the environment during all modes of operation.

In addition, the operation of the ESF exhaust filtration unit maintains a slightly negative pressure in portions of the building with respect to the environment following a LOCA and loss of offsite power, thus reducing the radioactive effluent released to the environment. Each of the four ESF exhaust filtration units consists of a demister, heater, HEPA filter, iodine adsorber, HEPA filter and fan, and complies with the environmental and components design criteria, the qualification testing provisions, and the intent of systems design criteria promulgated in NRC Regulatory Guide 1.52 as discussed in Section 1A(B). The removal of heat from ESF pump rooms during these periods is accomplished by using emergency fan coil units which maintain the rooms below the maximum environmental qualification temperature (122°F) though the temperature may rise to 129°F for a short duration. Ventilation system equipment is designed to operate in and to maintain the required ambient conditions.

The supply and exhaust ventilation units are a part of the modular arrangement for primary plant (controlled access area) ventilation systems, as shown on Figure 9.4-9 and as described in Section 9.4.3.

Ambient temperatures throughout the building during normal operation are presented in Table 9.4-2.

ESF exhaust fans are automatically energized from their Class 1E bus following a LOCA in either Unit 1 or Unit 2; thus a slightly negative pressure is provided within the Auxiliary, Fuel Handling and portions of the Safeguards Buildings. During a loss of offsite power at least one or more non-ESF exhaust fan(s) may be manually energized as required after 120 seconds (See Table 8.3-2) to achieve and maintain the slight negative pressure.

The system is provided with sufficient redundancy in equipment and power supplies to enable it to sustain a single active component failure without loss of function. Fire dampers in the exhaust duct are seismically qualified to remain open during an SSE. The redundant modular ESF exhaust fans are powered from redundant electric power trains.

To remove the large quantities of heat which are rejected from the motor driven ESF pumps following a DBA, each of the pump compartments is equipped with emergency fan coil units. Each emergency fan coil unit is powered from the same Class 1E bus as the equipment it serves. Also, each unit is connected in such a way as to start with the equipment it serves. The auxiliary cooling units are supplied with chilled water from the safety-related chilled water system shown on Figure 9.4-12 and described in Subsection 9.4F.

The system is provided with sufficient instrumentation to enable the operator to continuously monitor the system performance and manually switch to the standby units when required.

Exhaust air from the Safeguards Building is monitored by an in-line radiation monitor to assess its activity level and is passed through iodine adsorber beds prior to discharge to the atmosphere.

Portions of the exhaust system and all the emergency fan coil units are seismic Category I and ANS Safety Class 3 design. The main air supply system is non-nuclear safety class and seismic Category II. Also see Subsection 9.4.3 for ventilation of ESF equipment.

9.4.5.2 System Description

The Safeguards Building ventilation system is shown schematically on Figure 9.4-4. The primary plant air supply system delivers filtered and tempered outside air to each floor of the safety feature area through a duct distribution system. Each supply unit consists of a roughing filter, heating and cooling coils, and fan. The arrangement of filters and HVAC equipment is in accordance with the requirements of NRC Regulatory Guide 8.8, Revision 2, to achieve ALARA radiation levels. Chilled water is supplied to the cooling coils from the plant ventilation chilled water system as shown by Figure 9.4-11. This system provides chilled water only during normal operation.

Each of the four ESF exhaust filtration units consists of a demister, heater, HEPA filter, iodine adsorber, HEPA filter and fan, and complies with the environmental and components design criteria, the qualification testing provisions, and the intent of systems design criteria promulgated in NRC Regulatory Guide 1.52 as discussed in Section 1A(B).

The supply and exhaust system ductwork branches to all areas within the safety feature area. Wherever possible, airflow is directed from areas of lower potential radioactivity to areas of higher potential radioactivity. Unit heaters are provided in specific areas for operator comfort and to maintain ambient temperature above the lower limits for some process piping.

Emergency fan coil units, each of which comprises a water cooling coil and fan section, are provided for all compartments which contain motor driven ESF pumps and the electrical areas. The cooling medium for these fan coil units is chilled water, which is supplied by the nuclear safety chilled water system.

The emergency fan coil units are operated in conjunction with the equipment that they cool.

9.4.5.3 Safety Evaluation

The reliability and safety of the ESF area ventilation system is ensured by the following features:

1. The modular arrangement as shown on Figure 9.4-9 (with excess capacity and redundant power supplies) enables the system to sustain a loss of a single fan without loss of function. Standby fans can be remotely actuated from the Control Room. Each motor driven ESF pump room has a 100-percent-capacity cooling unit to remove the heat dissipated by the equipment.

2. Emergency fan coil units are supplied with chilled water from the safety chilled water system. Each chilled water system train and the auxiliary cooling units it serves are powered from the same train Class 1E bus.

3. Instrumentation and controls which incorporate audible and visual alarms in the Control Room facilitate continuous monitoring or system performance and alert the operator if the system malfunctions.

4. Failure modes for isolation valves and dampers are set so that their failure does not render the system inoperable. The appropriate safeguards building exhaust dampers (Figure 9.4-4 Sh. C) are abandoned in place in the OPEN position.

5. The cooling units, and safety features chilled water system are of seismic Category I and ANS Safety Class 3 design. The exhaust system is of ANS Safety Class 3, seismic Category I, and NNS, seismic Category II, design. The supply system components are designed to appropriate seismic criteria, where necessary, to negate the possibility of these components interfering with the operation of safety-related components.

6. Exhaust air from portions of the Safeguards Building is passed through iodine adsorber beds prior to its discharge. Data concerning the ESF and non-ESF iodine adsorbers are presented in Table 9.4-6.

7. In case of a DBA and subsequent operation of the ECCS, the emergency fan coil units operate on a recirculation basis and provide cooling of the ECCS pumps.

8. Fire dampers in the exhaust duct which are Seismic Category II are qualified to remain open during SSE.

9. Exhaust air from portions of the Safeguards Building is monitored by an in-line radiation monitor to assess its activity level and is passed through iodine adsorber beds prior to discharge to the atmosphere.

9.4.5.4 Inspection and Testing Requirements

1. See Subsection 9.4.2.4.

2. The HEPA and iodine adsorbers of the ESF Units are tested periodically during plant operation in accordance with NRC R.G 1.52 (See appendix 1A(B)). Filter Units are arranged to facilitate filter replacement.

REFERENCES

1. 10 CFR Part 50, Appendix A, GDC 2, Design Basis for Protection Against Natural Phenomena.

2. 10 CFR Part 50, Appendix A, GDC 4, Environmental and Missile Design Bases.

3. 10 CFR Part 50, Appendix A, GDC 5, Sharing of Structures, Systems, and Components.

4. 10 CFR Part 50, Appendix A, GDC 19, Control Room.

5. 10 CFR Part 50, Appendix A, GDC 60, Control of Releases of Radioactive Materials to the Environment.

6. 10 CFR Part 50, Appendix A, GDC 64, Monitoring Radioactivity Releases.

7. NRC Regulatory Guide 1.26, Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants, Revision 3, February 1976, U.S. Nuclear Regulatory Commission.

8. NRC Regulatory Guide 1.29, Seismic Design Classification, Revision 2, February 1976, U.S. Nuclear Regulatory Commission.

9. NRC Regulatory Guide 1.13, Fuel Storage Facility Design Basis, Revision 1, December 1975, U.S. Nuclear Regulatory Commission.

10. NRC Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants, Revision 1, July 1976, U.S. Nuclear Regulatory Commission.

11. NRC Regulatory Guide 1.78, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, June 1974, U.S. Nuclear Regulatory Commission.

12. NRC Regulatory Guide 1.95, Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release, Revision 1, January 1977, U.S. Nuclear Regulatory Commission.

13. ANSI N510 1975, Testing of Nuclear Air Cleaning Systems.

14. ANSI N509 1976, Nuclear Power Plant Air Cleaning Units and Components.

15. International Report NAA-SR-100, Leakage Characteristics of Openings for Reactor Housing Components.

16. American Air Filter Topical Report AAF-TR-7101, Design and Testing of Fan Cooler Filter Systems for Nuclear Applications, February 1972.

17. Test Code for Air Moving Devices, Air Moving and Conditioning Association (AMCA) 210 67.

18. ANSI N101.1 1972, Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles.

19. American Society of Heating, Refrigeration, and Air Conditioning Engineers, Guide and Data Handbooks Vol. I-IV.

TABLE 9.4-1
DESIGN CONDITIONS - OUTDOORS

Summer Design Temperature, °F	102 DB/80 WB
Extreme Summer Temperature, °F[(a)]	113 DB/79 WB
Winter Design Temperature, °F	20
Extreme Winter Temperature, °F	4
Annual Average Temperature, °F	66 DB
Daily Average Temperature Range, °F	20 DB
Latitude	32
Elevation, Ft.	810
Average Wind Velocity, mph (Approximately)	15

a) This value was utilized to evaluate the impact of extreme outdoor weather condition on safety related HVAC System. This is not a design condition.

TABLE 9.4-2
DESIGN CONDITIONS - INDOOR

(Sheet 1 of 4)

Building or Area		Normal Plant Conditions		
		Maximum DB(°F)	Minimum DB(°F)[(a)]	Relative Humidity (%)
Auxiliary Bldg. (AB)				
1.	Elevator Machine Room	122	40	[(b)]
2.	Heat Exchanger Area (el. 790'-6")	122	40	[(b)]
3.	Valve & Piping Area (el. 790'-6")	122	40	[(b)]
4.	Operating Valve Rooms (el. 822')	122	40	[(b)]
5.	Auxiliary Steam Drain Tank Equipment Room (el. 790'-6")	122	40	[(b)]
6.	Valve Rooms (el. 810'-6" and 831'-6")	122	40	[(b)]
7.	Operating Valve Room (el. 862'-6")	122	40	[(b)]
8.	ESF Pump Rooms	122	40	[(b)]
9.	Boric Acid Storage Tank Rooms	104	65	[(b)]
10.	All other areas	104	40	[(b)]
Electrical and Control Bldg. (ECB)				
1.	Control Room	80	70	35-50
2.	Mechanical Equipment Rooms	104	40	[(b)]
3.	UPS & Distribution Rooms	104	40	[(b)]
4.	Uncontrolled Access Area	104	40	[(b)]
5.	Battery Rooms	104	70	[(b)]
6.	Chiller Equipment Areas (el. 778')	122	40	[(b)]
7.	All other Areas	104	40	[(b)]

TABLE 9.4-2
DESIGN CONDITIONS - INDOOR

(Sheet 2 of 4)

Building or Area	Normal Plant Conditions		
	Maximum DB(°F)	Minimum DB(°F)[a]	Relative Humidity (%)
Fuel Handling Building			
1. Stairs (el. 810'-6" & 841')	122	40	(b)
2. Spent Fuel Pool Cooling Heat Exchanger & Pump Rooms	122	40	(b)
3. New Fuel Storage Pit	122	40	(b)
4. All other Areas	104	40	(b)
Reactor Containment Bldg. (RCB)			
1. Outside Missile Barrier	120	50	(b)
2. Inside Missile Barrier	120	50	(b)
3. CRDM Shroud (Air Temperature)	163 (CRDM Air Handling Unit inlet) (Unit 1) 163 (outlet) (Unit 2) 120 (inlet)	50 50	(b) (b)
4. CRDM Platform	140	50	(b)
5. Detector Well and Reactor Cavity	135[c]	50	(b)
6. Reactor Coolant Pipe Penetrations	200	50	(b)
Safeguards Bldg. (SGB)			
1. MS & FW Piping Area	104	40	(b)
2. Valve Isolation Tank Rooms (el.790'-6")	122	40	(b)

TABLE 9.4-2
DESIGN CONDITIONS - INDOOR

(Sheet 3 of 4)

Building or Area	Normal Plant Conditions		
	Maximum DB(°F)	Minimum DB(°F)[a]	Relative Humidity (%)
3. Valve Rooms (el. 790'-6")	122	40	[b]
4. RHR/Cont. Heat Exchanger Rooms (el. 790'-6")	122	40	[b]
5. ESF Pump Rooms	122	40	[b]
6. All Other Areas	104	40	[b]
Diesel Generator Bldg. (DGB)			
1. Diesel Generator Rooms	122	40[d]	[b]
2. Day Tank	122	40[d]	[b]
3. Equipment Rooms	130	40[d]	[b]
Service Water Intake Structure (SWIS)			
1. SWIS - All Areas	127	40	[b]
Turbine Bldg. (TB)			
1. Switchgear Area	104	40	[b]
2. ERF-Computer Battery Room	104	40	[b]
3. CAS Room, CPU & UPS Rooms	75	68	35-50
4. Battery Rooms & HVAC Equipment Rooms	104	40	[b]
5. Office and Service Area A/C	80	68	[b]
6. Laboratories	80	68	35-50
7. Alternate RCA Access-Unit 2	80	68	[b]
8. Tool Room	80	65	[b]
9. All Other Areas	122	40	[b]

TABLE 9.4-2
DESIGN CONDITIONS - INDOOR

(Sheet 4 of 4)

Building or Area		Normal Plant Conditions		
		Maximum DB(°F)	Minimum DB(°F)[a]	Relative Humidity (%)
Miscellaneous Building (MB)				
1.	Alternate Access Point (AAP) Bldg.	80	68	(b)
2.	Circulating Water Intake Structure Chlorination Bldg.	115	40	(b)
3.	SWIS - Chlorination Building	115	40	(b)
4.	Switchyard Relay House	80	68	(b)
5.	Security Office	80	68	(b)
6.	Battery Rooms on Turbine Building Deck	87	67	(b)

a) The winter minimum design temperature in non-air conditioned areas, including Diesel Generator Area, Electrical Area and Service Water Intake Structure of 40oF. Except for the Diesel Generator Area, this temperature is maintained by the HVAC systems without considering equipment heat released during startup, normal plant operation, hot standby and refueling operation, except for the Electrical Area Ventilation System (see Section 9.4C.3). There is no heating from the ventilation systems available during loss of offsite power. The heat released by safety-related equipment is sufficient to keep the temperature of the room above the freezing point. The Boric Acid Storage Tanks contain concentrated boric acid solution. To preclude recrystalization, these tanks are required to be located in a room maintained at 65°F or higher.

b) Uncontrolled

c) Occasional temperature spikes of 175°F DB may occur. The reactor cavity exhaust temperature is normally 15 degrees higher (e.g. 150°F and 190°F respectively).

d) Temperatures in the Diesel Generator Area are maintained with the help of the Diesel keep warm system.

TABLE 9.4-3
EQUIPMENT LINE-UP FOR CONTROL ROOM AIR-CONDITIONING SYSTEM
MODES OF OPERATION

Information shown on Figure 9.4-1.

TABLE 9.4-4
FILTER DESIGN PARAMETERS

(Sheet 1 of 2)

Prefilters	
Atmospheric Dust Spot Efficiency Rating per ASHRAE[(a)] Standard 52	80-95%
Initial (clean) resistance (pressure drop) at rated air flow	0.65 in. W.G. (Max)
Final (dirty) resistance (pressure drop) at rated air flow	1 to 2 in. W.G.
Rated Air Flow	1300 CFM/CELL at 0.4 in. W.G.
HEPA Filters	
Initial (clean) resistance (pressure drop) at rated air flow	1.0 in. W.G.
Final (dirty) resistance (pressure drop) at rated air flow	3.5 in. W.G.
Final (dirty) resistance (pressure drop)at rated air flow for Containment Preaccess Filtration Units	2.0 in. W.G.
Rated Air Flow	1500 CFM/CELL at 1.2 in. W.G.
Filtration Efficiency: Penetration through complete filter (medium, frame and gasket) when operated at rated capacity and tested with thermally generated DOP[(b)] of uniform 0.3 micron droplet size, in accordance with Edgewood Arsenal Manual.	a) less than 0.05% (For Control Room HVAC Filtration and Recirculation Units) b) less than 1.0% (For all other Filtration Units)
Iodine Adsorber	
Material	Impregnated coconut charcoal (8 x 16 mesh granules)

TABLE 9.4-4
FILTER DESIGN PARAMETERS

(Sheet 2 of 2)

Efficiency (Assigned)	a)	Engineered Safety Feature (ESF) (R.G. 1.52)
		Primary Plant Filtration Units 95%
		Control Room HVAC Filtration and Recirculation Units 99%
	b)	Non-ESF (Normal Ventilation Air Filtration and Adsorption Units) (R.G. 1.140)
		Primary Plant Filtration Units 90%
		Hydrogen Purge Filtration Units 90%
		Containment Preaccess Filtration Units 90%
Leakage acceptance test requirement		R-11 (or approved alternate)
Penetration, % allowable for in place tests.		0.05% (For Control Room HVAC Units), 1.00% (For all other Filtration Units)
Average Residence Time		0.25 sec per 2" of adsorbent bed
Maximum Permissible Loading		2.5 mg of total Iodine (radioactive plus stable) per gram of activated carbon

a) American Society of Heating Refrigerating and Air-Conditioning Engineers

b) Dioctyl Phthalate

TABLE 9.4-5
CONTAINMENT AIR RECIRCULATION UNIT DATA

(Sheet 1 of 2)

Number installed	four
Number required to operate	three
Type of coil	Chilled water
Fan capacity, ft^3/min per cooling unit	65,000
Conditions	
Containment temperature, °F (maximum)	120
Total pressure, psig	Atmospheric
Cooling medium	Ventilation chilled water system

Neutron Detector Well Unit Data

Number installed	two
Number required to operate	one
Type of coil	Chilled water
Fan capacity, ft^3/min per cooling unit	13,100
Temperature, °F (average)	150 (reactor cavity exhaust)
Cooling medium	Ventilation chilled water system

Control Rod Shroud Ventilation Unit Data

Number installed	two
Number required to operate	one
Type of Coil	Chilled Water
Fan capacity, ft^3/min per fan unit	Approximately 60,000

TABLE 9.4-5
CONTAINMENT AIR RECIRCULATION UNIT DATA

(Sheet 2 of 2)

Temperature, °F (outlet air average)	163 maximum
Cooling Medium	Ventilation Chilled Water System

Containment Preaccess Filtration Subsystem

Number installed	two
Number required to operate	two
Fan capacity, ft^3/min per filtration unit	Approximately 15,000
Temperature, °F (air average)	120 maximum

Normal Containment Purge Supply and Exhaust Subsystem

Number installed (ppv supply/exhaust units)	one/two
Number required to operate	one/two
Fan capacity, f^3/min per fan unit	30,000/15,000
Temperature, °F (air average)	60

TABLE 9.4-6
IODINE ADSORBERS

Filter	Bed Depth (in. thick)	Residence Time (sec)	Temperature (°F)	Inlet Humidity (Percent)
Preaccess filtration, two 15,000 ft^3/min units	4	0.50	120 (Max.)	(a)
Controlled access air cleanup, twelve 15,000 ft^3/min units	4	0.50	120 (Max.)	(a)
Controlled access ESF air cleanup, four 15,000 ft^3/min units	4	0.50	155 (Max.)	70 (Max.)
Hydrogen purge, two 1000 ft^3/min units (one standby)	4	0.50	160 (Max.)	70 (Max.)
Control room emergency pressurization, two 800 ft^3/min units (one standby)	4	0.50	120 (Max.)	70 (Max.)
Control Room emergency filtration, two 8000 ft^3/min units (one standby)	4	0.50	86 (Max.)	(b)

a) Humidity not controlled.

b) Controlled by Control Room A/C units

TABLE 9.4-7
EFFECT OF FUEL POOL EVAPORATION AS A FUNCTION OF WATER TEMPERATURE

Air Temperature (R)	Air Velocity (ft/min)	Pool Water Tempurature (F)	Air Relative Humidity (Percent)
115	200	120	58
115	200	130	68
115	200	140	80
115	200	150	97
115	200	160	100
115	200	170	100
115	200	180	100

TABLE 9.4-8
FAILURE MODE AND EFFECT ANALYSIS OF CONTROL ROOM HVAC SYSTEM

(Sheet 1 of 3)

Component	Malfunction	Effect on System Operation	Comments
1. Control Room air conditioning units	a. Unit stops. b. Unit fails to start.	No effect	Sufficient redundancy provided for 100-percent standby feature.
2. Control Room makeup air supply fan units	a. Fan stops. b. Fan fails to start	No effect	Two 100-percent-capacity makeup air supply fan units are provided, one operating and one standby.
3. Control Room exhaust fan units	a. Fan stops. b. Fan fails to start.	No effect	Two 100-percent-capacity exhaust fan units are provided, one one operating and one standby.
4. Control Room kitchen and toilet exhaust fan units	a. Fan stops. b. Fan fails to start.	No effect	Two 100-percent-capacity kitchen and toilet exhaust fan units are provided, one operating and one standby.
5. Control Room emergency pressurization air supply fan units	a. Fan stops. b. Fan fails to start	No effect	Two 100-percent capacity emergency pressurization air supply fan units are provided, one operating and one standby.
6. Control Room emergency filtration system	Filter train is clogged or inoperative.	No effect	Two 100-percent-capacity emergency filtration systems are provided, one operating and one standby.

TABLE 9.4-8
FAILURE MODE AND EFFECT ANALYSIS OF CONTROL ROOM HVAC SYSTEM

(Sheet 2 of 3)

Component		Malfunction	Effect on System Operation	Comments
7.	Control Room emergency pressurization filtration system	Filter train is clogged or inoperative.	No effect	Two 100-percent-capacity emergency pressurization filtration systems are provided, one operating and one standby.
8.	Control Room HVAC system isolation and modulating dampers	Fails to operate properly	No effect	Failure position of dampers will not impede operation of the components listed above.
9.	Radiation detectors	Fails to operate	No effect	Control Room HVAC system automatically switches to emergency recirculation if any one of the detectors fail to operate.
10.	Control Room emergency pressurization air-conditioning system	Fire isolated.	No effect	Effected air-conditioning system is Two 100-percent-capacity filtration systems are provided, one operating and one standby.
11.	Control Room emergency filtration air-conditioning system	Fire isolated.	No effect	Effected air-conditioning system is Two 100-percent-capacity filtration systems are provided, one operating and one standby.

TABLE 9.4-8
FAILURE MODE AND EFFECT ANALYSIS OF CONTROL ROOM HVAC SYSTEM

(Sheet 3 of 3)

Component		Malfunction	Effect on System Operation	Comments
12.	Electric control	Power failure	No effect	Redundant controls, powered from redundant Class 1E power sources, are provided.
13.	HVAC dampers	Air failure	No effect	Air accumulators are provided on dampers CPX-VADPOU-14, 15, all others fail in the position required to operate in Emergency Recirculation Mode.
14.	Fire in an area of the Control Room complex[a]		No effect	Fire dampers are provided which isolate the affected area, preventing the spread of fire.

a) Low probability of electrical equipment fire.
Control Room complex is provided with fire protection equipment (portable chemical extinguishers).

TABLE 9.4-9
FAILURE MODE AND EFFECT ANALYSIS OF FUEL HANDLING AREA HVAC SYSTEM

(Sheet 1 of 2)

Component	Malfunction	Effect on System Operation	Comments
1. Fuel handling area modular supply units	a. Fan fails to start. b. Fan stops. c. Cooling and heating coils fail to operate properly.	No effect	The modular supply arrangement provides sufficient redundancy in order to maintain perferred air flow patterns.
2. Primary plant exhaust fans	a. Fan fails to start b. Fan stops. c. Filter train clogged.	No effect	The modular exhaust arrangement provides sufficient redundancy in order to maintain preferred air flow patterns.
3. Primary plant exhaust	Loss of fuel pool cooling	No effect	Increased humidity will not affect PPV exhaust unit charcoal because of mixing of exhaust air from safeguard and auxiliary buildings.
4. Primary plant ventilation system	Loss of offsite power	No effect	Emergency fan coil units are provided for Spent Fuel Pool Cooling Pump rooms. Concrete heat sinks are adequate for heat dissipation until ventilation is restored.

TABLE 9.4-9
FAILURE MODE AND EFFECT ANALYSIS OF FUEL HANDLING AREA HVAC SYSTEM

(Sheet 2 of 2)

Component	Malfunction	Effect on System Operation	Comments
5. Spent fuel pool area fans	a. Fan fails to start. b. Fan stops.	No effect	Two 50-percent-capacity fuel pool fans are provided. Equipment not required for safe shutdown or fuel handling accident.
6. Spent fuel pool area electric heaters	Fails to operate	No effect	The rate of evaporation from fuel pool will decrease.
7. Spent fuel pool demister	Decrease in efficiency	No effect	During fuel handling this air is exhausted through the Primary Plant Ventilation exhaust system.

TABLE 9.4-10
FAILURE MODE AND EFFECT ANALYSIS OF SAFETY RELATED DAMPERS IN HVAC SYSTEMS

(Sheet 1 of 3)

BUILDING	FUNCTION & TYPE OF DAMPERS	MALFUNCTIONS	FAILURE MODE	EFFECT ON SYSTEM OPERATION	COMMENTS
1. Fuel, Safeguards and Auxiliary Buildings	a. Room isolation dampers	1. Fail to operate properly 2. Instrument air fails 3. Loss of offsite power.	Fail Closed (F.C.)	No effect	Failure mode of damper is such that it will not impede the operation of the system. Where required, the dampers are locked open.
	b. Not used				
	c. Non-ESF filtration systems isolation dampers: 1. Opposed blade type 2. Gravity type	1. Fail to operate properly 2. Instrument air fails. 3. Loss of offsite power.	See Fig. 9.4-9	No effect	Units are provided each with its own set of dampers. The opposed blade dampers are operated manually.
	d. ESF Filtration System isolation dampers: 1. Opposed blade type 2. Gravity type	1. Fail to operate properly. 2. Instrument air failure. 3. Loss of offsite power.	F.O. for ESF filter units	No effect	Failure mode of damper is such that it will not impede the system operation. Units are provided each with its own set of dampers. The opposed blade dampers can be operated manually if required.
2. Control Room	a. Outside air inlet bubble tight type dampers CPX-VADPOU 14 & 15	1. Fail to operate properly. 2. Instrument air fails.	F.C.	No effect	Failure mode of damper will not impede the system operation. Two-100% redundant outside air intakes each with its own damper are provided. Air accumulators are provided.

TABLE 9.4-10
FAILURE MODE AND EFFECT ANALYSIS OF SAFETY RELATED DAMPERS IN HVAC SYSTEMS

(Sheet 2 of 3)

BUILDING	FUNCTION & TYPE OF DAMPERS	MALFUNCTIONS	FAILURE MODE	EFFECT ON SYSTEM OPERATION	COMMENTS
	b. Control Room exhaust and toilet exhaust fans isolation dampers: 1. Opposed blade type 2. Gravity type CPX-VADPMU-05, 06 CPX-VADPOU-27, 28 CPX-VADPGU-28, 29, 30, 31	1. Fail to operate properly. 2. Power failure to motor operated dampers. 3. Gravity dampers not operable. 4. Loss of offsite power.	Failure Mode Fail As Is (FAI)	No effect	Failure mode of dampers will not impede the system operation. Two-100% capacity fan units each unit with its own set of dampers are provided. Motorized dampers are powered from emergency power supply.
	c. Emergency filtration and emergency pressurization filtration unit and fans isolation 1. Opposed Blade type CPX-VADPOU-41 & 42 2. Gravity type CPX-VADPGU-01, 02, 03, 04 3 Bubble Tight type CPX-VADPOU-22 & 23	1. Fail to operate properly 2. Power failure to motor operated dampers. 3. Gravity dampers not operable. 4. Loss of offsite power.	CPX-VADPOU--41, 42 F.O. CPX-VADPOU-22, 23-FAI	No effect	Motored dampers are powered from emergency power supply. Failure mode of dampers will not impede the system operation. Two-100% capacity fan and filtration units with its own set of dampers.
	d. Air conditioning units isolation dampers: 1. Opposed blade type 2. Gravity type CPX-VADPOU-10, 11, 12, 13 CPX-VADPOU-5, 6, 17, 18	1. Fail to operate properly. 2. Instrument air fails. 3. Gravity dampers not operable. 4. Loss of offsite power.	F.O.	No effect	Four-50% capacity air conditioning units with its own set of dampers are provided. Two units are operating and two units are on standby. Failure mode of dampers will not impede the operation of the system.

TABLE 9.4-10
FAILURE MODE AND EFFECT ANALYSIS OF SAFETY RELATED DAMPERS IN HVAC SYSTEMS

(Sheet 3 of 3)

BUILDING	FUNCTION & TYPE OF DAMPERS	MALFUNCTIONS	FAILURE MODE	EFFECT ON SYSTEM OPERATION	COMMENTS
	e. Makeup supply air fans isolation dampers opposed blade type CPX-VADPOU-16 & 19	1. Bubble Tight type CPX-VADPOU-16 & 19 2. Power failure to motor operated dampers. 3. Instrument air 4. Loss of offsite power.		No effect	Two-100% capacity fan units its own damper set, one unit operating and one on standby. Failure mode of dampers will not impede the operation on standby. Failure mode of dampers will not impede the operation of the system. Power to motor operated dampers is supplied from emergency power. Inlet bubble tight dampers provide redundant isolation.
	f. Damper in recirculation lines opposed blade type CPX-VADPOU-25	1. Fail to operate properly 2. Instrument air fails 3. Loss of offsite power.	FC	No effect	Failure mode of damper is such that it will not impede the operation of the system. It is desired to fail closed and that is the safe mode of failure during accident conditions. Damper can be operated manually if required.

APPENDIX 9.4A - CONTAINMENT VENTILATION SYSTEMS

Separate ventilation systems are furnished for the Containments of Units 1 and 2. The Containment ventilation systems, as shown on Figure 9.4-5, consist of the Containment air recirculation and cooling Control Rod Drive Mechanism (CRDM) ventilation, neutron detector well cooling, reactor coolant pipe penetration cooling, preaccess filtration, Containment pressure relief, and Containment purging systems. Specific data for the various systems are presented in Table 9.4-5.

9.4A.1 DESIGN BASES

9.4A.1.1 Containment Air Recirculation and Cooling System

The environment within containment is controlled by the containment recirculation and cooling system during startup, power operation and hot standby.

The Containment air recirculation and cooling system maintains Containment ambient temperature at or below 120°F during normal plant operation. The Containment air recirculation and cooling system together with the CRDM ventilation system is designed to remove all heat generated by equipment within the Containment, with the exception of the reactor coolant pump motors, which are cooled separately by the CCWS.

The system operates during a loss of offsite power and provides mixing of fresh air during refueling and shutdown, but it is not operative following a DBA. Postaccident cooling is provided by the Containment Spray System (CSS). (See Section 6.2.2.)

9.4A.1.2 Control Rod Drive Mechanism Ventilation System

The CRDM ventilation system maintains temperatures within the CRDM shrouds in accordance with NSSS requirements. The CRDM cooling coils supplement the general containment cooling provided by the Containment Air Recirculation and cooling system. Specific data concerning this system are presented in Table 9.4-5.

9.4A.1.3 Neutron Detector Well Cooling System

Neutron detector well cooling units prevent neutron detectors from exceeding their temperature limitations. (See Table 9.4-5.) These units also provide cooling for reactor shield wall concrete and nozzle supports. Two 100-percent capacity cooling units are provided for redundancy, and cooling water is provided by the plant ventilation chilled water system.

9.4A.1.4 Containment Preaccess Filtration

The need for safe, periodic, or emergency access to the Containment necessitates the use of Containment preaccess filtration. This filtration system provides air circulation and filtering throughout all Containment areas. Containment preaccess filtration equipment reduces the concentration of fission product particulate activity in the containment atmosphere prior to personnel entering the Containment. Iodine adsorbers are designed to reduce the airborne halogen activity. The design is based on estimates of reactor coolant leakage supplied by the NSSS manufacturers. The reduced level of airborne particulate by HEPA filters permits access

to the Containment for brief time intervals, without exceeding the limitation of 100 mrem/week. The containment preaccess filtration system minimizes the need for containment purging.

9.4A.1.5 Containment Purge Supply and Exhaust System

Containment purge supply and exhaust equipment satisfies the prerequisites for safe, prolonged access to the Containment following shutdowns.

The system is utilized during shutdown only; it is not used during startup, power or hot standby operations. Since the system will not be used during power operations, its operation is not considered in the ECCS backpressure computation presented in Section 15.6.5.

Tempered fresh air is supplied and circulated throughout the Containment and ventilation equipment is designed to maintain Containment temperature above 50°F during the winter season. (The supply and exhaust filtration units are a part of the modular arrangement for the primary plant ventilation system as shown on Figure 9.4-9.) Exhaust air is passed through an exhaust filtration unit which consists of a prefilter, two HEPA filters, and an iodine adsorber and discharged into the atmosphere through the plant ventilation vent. The rate of release ensures that the offsite concentration of radioactive material is within the limits of 10 CFR Part 20. A discussion of the estimated annual radioactivity discharges from purging is presented in Section 11.1. Specific data on the various systems are shown in Table 9.4-5.

9.4A.1.6 Containment Pressure Relief System

The pressure relief system is used intermittently during startup, power or hot standby operations. The pressure relief system is the only system that can provide an open path from the containment atmosphere to the environs during startup, power or hot standby conditions, and is designed in accordance with Branch Technical Position CSB 6-4. A 3-inch restrictive flow orifice plate is provided so that the containment pressure relief system may be open continuously.

The Containment pressure relief system is designed to relieve excess Containment pressures caused by temperature transients or air leakage from pneumatic actuators during normal operation. The Containment pressure relief system is connected to the Containment purge exhaust duct outside Containment, downstream of both system Containment isolation valves and upstream from the primary plant exhaust filters. This connection ensures that any air from the Containment is filtered prior to release to the atmosphere, via the plant ventilation vent.

The Containment pressure relief system is designed to reduce the Containment pressure from a maximum of 1.5 psig to atmospheric pressure.

9.4A.1.7 Reactor Coolant Pipe Penetration Cooling System

The reactor coolant pipe penetration cooling in primary shield wall system is designed to dissipate heat conducted to the supports by the reactor coolant lines utilizing forced convection to prevent dehydration of concrete.

9.4A.2 SYSTEM DESCRIPTION

9.4A.2.1 Containment Recirculation and Cooling System

During normal operation, air in the Containment is recirculated and maintained at or below 120°F. Four cooling units are provided, each sized for 33 1/3 percent (65,000 scfm) of the normal duty cooling load. Standby units are manually actuated on receipt of a failure alarm in the Control Room. Fans are provided with a connection to the Class 1E buses to preclude loss of cooling caused by loss of offsite power. The cooling medium is water provided from the plant ventilation chilled water supply during normal operation and loss of offsite power, because the plant ventilation chillers are also supplied from the Class 1E buses during loss of offsite power conditions. Instruments are provided to continuously monitor air temperature, humidity, and pressure within the Containment during all phases of reactor operation.

The Containment recirculation units together with the CRDM ventilation system remove the entire Containment heat load during normal operation and loss of offsite power conditions including the CRDM heat load. The neutron detector well heat load is removed by a separate set of coils.

The cooled air is distributed throughout the Containment by the supply ductwork with no return ductwork provided. Instead, the warm air rises through various openings in the floors and returns to the suction side of the fan coil units.

9.4A.2.2 Control Rod Drive Mechanism Ventilation System

Containment air is drawn through the CRDM shrouds and through the CRDM cooling coils and discharged to the containment atmosphere. Two 100-percent capacity fans are provided. The standby unit, isolated by dampers in the duct, is manually actuated on receipt of a failure alarm in the Control Room. In the event normal power supplies are interrupted and the reactor is maintained at hot standby, the CRDM fans are operated from the emergency power supply. The cooling medium to the CRDM cooling coils is provided by the plant ventilation chilled water system. The CRDM ventilation system does not operate in the event of a DBA.

9.4A.2.3 Neutron Detector Well Cooling System

The neutron detector well and nozzle support cooling system is a closed loop system. The air is cooled, and directed up around the reactor from the underside of the vessel. Part of the air cools the neutron detectors, and part of the air is directed to the nozzle supports. The air is returned to the fan via piping embedded in the concrete. Two 100-percent capacity cooling units are provided with each consisting of a fan, two cooling coils, and inlet and outlet dampers. Both units are connected to redundant emergency diesel generator buses to preclude loss of cooling in case of a loss of offsite power. A temperature monitoring device continuously monitors air temperature in the return air ducts of the detector well system.

9.4A.2.4 Containment Preaccess Filtration

Preaccess filtration equipment can be operated prior to reactor shutdown for Containment access. Air inside the Containment is recirculated by two 50-percent capacity filtration units. Each filtration unit includes a fan, a roughing filter, two HEPA filters, and one iodine adsorber. Both units are required to operate for the design iodine removal capacity. The equipment is designed to remove fission product iodine gas as well as radioactive particles to permit prompt

access to the Containment. Ductwork connected to the suction side of the preaccess filtration unit extends to all levels of the Containment to ensure that the containment atmosphere is uniformly decontaminated. Air is discharged at the suction side of the Containment recirculation system, which supplies the air through ductwork and guarantees proper mixing. The minimum total airflow with clean filters is 30,000 ft3/min, with an iodine adsorber having an assigned radioiodine removal efficiency in compliance with Regulatory Guide 1.140 (See Appendix 1A(B)). Filters are also designed in accordance with the criteria of Regulatory Guide 1.140. Charcoal adsorbers are provided with a fire protection water deluge system. The system operation is described in section 9.5.1.4.2.2.b.2. Data concerning preaccess filtration equipment are presented in Tables 9.4-6, 9.4-4, and 9.4-5.

9.4A.2.5 Containment Purge Supply and Exhaust System

The purge supply and exhaust units are part of the primary plant ventilation systems modular arrangement (see Figure 9.4-9). Description of each system is as follows:

1. Containment Purge Supply

   Fresh, outside air is passed through a filter and heating and cooling coils and discharged near the recirculation supply fans in the Containment. Thermostatically controlled electric heating coils are provided in the purge supply duct to maintain minimum temperatures for personnel comfort during shutdown.

   One 30,000 scfm filter heating and cooling coil plenum and a fan are required. The containment purge supply isolation valves are designed to close automatically on detection of high radiation levels by the containment air radiation monitor or on a Phase A isolation signal as discussed in sections 6.2.4 and 7.3.

2. Containment Purge Exhaust

   Exhaust air is drawn through a prefilter, two HEPA filters, and an iodine adsorber and exhaust fan, and released to the atmosphere via the plant ventilation vent.

   Two filter plenums with associated fans are required to maintain design purge rate. The air is monitored before discharge to the environment, to limit the concentration of contaminants as required by 10 CFR Part 20. The containment purge exhaust isolation valves are designed to close automatically on detection of high radiation levels by the Containment air radiation monitor or on a phase A isolation signal as discussed in Sections 6.2.4 and 7.3.

9.4A.2.6 Containment Pressure Relief

The Containment pressure relief system is manually operated from the Control Room. A high Containment pressure alarm, located in the Control Room, alerts the operators to pressures exceeding normal operating limits. The alarm has a set point dictated by the maximum pressure permitted within the Containment during normal operation.

The pressure relief line, as shown on Figure 9.4-6, is connected to the Containment purge exhaust system; the containment pressure relief system requires only one penetration. The

exhaust air is decontaminated by the primary plant modular filtering trains prior to being discharged to the plant ventilation discharge duct.

The only potential leakage path from the containment building in existence during power operation is the 18 in. in diameter pressure relief line, and it will only open to the atmosphere if it has been manually actuated in response to compartment pressurization. Since it is anticipated that the line will be closed most of the time, it is not included in the ECCS backpressure analysis presented in Section 15.6.5. Moreover, even if the pressure relief line were open, it would have an insignificant effect on the pressure calculated during the core reflood transient because fast-closing isolation valves are installed.

Both pressure relief Containment isolation valves automatically close if high radiation is detected by the containment particulate-iodine-gaseous monitor, on a phase A isolation signal, or on containment ventilation isolation, as discussed in Sections 6.2.4 and 7.3. The isolation valves are 18 in. in diameter and will close within three seconds after receiving the signal.

The valve and actuator design bases include consideration or buildup of Containment pressure for the LOCA break spectrum and relief line flows as a function of time up to and during valve closure.

A debris screen is provided inboard of the inside containment isolation valve. The screen is designed to seismic category II criteria, and is placed in the flow path of the containment pressure relief system for protection of the isolation valves from debris.

Additional requirements for the design of the system isolation valves are presented in Section 6.2.4. Leak testing of containment pressure relief isolation valves is specifically addressed in Table 6.2.4-2.

9.4A.2.7 Reactor Coolant Pipe Penetration Cooling System

The Reactor Coolant Pipe Penetration Cooling System supplies high velocity cooled air to each of the reactor coolant hot cold leg pipe penetration supports. This cool air flow removes heat conducted to the supports by the pipe to keep the pipe tunnel concrete temperature below 200°F. The system consists of four fans (2000 scfm each), fan outlet isolation dampers, and ductwork. The system is divided into two subsystems: each subsystem having two 100-percent capacity fans supplying the four penetrations associated with two reactor coolant loops. During normal plant operation, two fans are operative, i.e., one in each subsystem. The fans are the centrifugal, direct drive type, each equipped with differential pressure indicating switches, three position remote manual hand switches located on the main control board in the Control Room, and parallel blade, counterweight operated gravity dampers at each outlet to prevent free wheeling of the standby fan. Standby fans are automatically actuated on receipt of a fan trip signal in the Control Room.

The Containment recirculation system supplies tempered air to the immediate area surrounding the fan inlets. Air is drawn in by the operating fan and supplied through ductwork directly to each pipe penetration. The warm air is forced out by the supply air and rises up, thus being cooled and recirculated by the Containment recirculation system. The ductwork is arranged with balancing dampers to ensure that equal amounts of air are supplied to the four supports it connects.

The Reactor Coolant Pipe Penetration Cooling System is not required to operate following a loss of offsite power or a DBA.

9.4A.3 SAFETY EVALUATION

The reliability and safety of the Containment ventilation systems is ensured by the following features:

1. Redundancy in equipment and power supplies with the exception of the preaccess filtration system enables the systems to sustain a single active component failure without total loss of function.

2. Instrumentation and controls which incorporate audible and visual alarms in the Control Room facilitate continuous monitoring of the system performances and alert the operator for system malfunctions.

3. Standby units can be remotely actuated from the Control Room manually.

4. Failure modes for equipment shutoff valves and dampers are set so that their failure does not render the system inoperable.

5. The systems inside the Containment are seismic Category II.

6. Penetrations are ANS Safety Class 2 and seismic Category I.

7. Exhaust air is passed through iodine adsorber beds prior to its discharge.

9.4A.4 INSPECTION AND TESTING REQUIREMENTS

1. Shop inspection and testing are performed for all equipment, including heating and cooling coils and controls.

2. The system is initially tested and adjusted for proper flow paths, flow capacities, heating and cooling capacities, mechanical operability, and filter efficiency.

3. Fans are rated and tested in accordance with the standards of the Air Moving and Conditions Association (AMCA).

4. Heating and cooling coils are tested in accordance with the manufacturer's standards. Ductwork is tested in accordance with industry standards.

5. Redundant standby equipment is operated on a cyclic basis to ensure the availability of the equipment.

6. The HEPA and charcoal adsorber of the containment filtration units are tested in accordance with NRC Reg. Guide 1.140 [See appendix 1A(B) under NRC Reg. 1.140], except for the following:

   The containment preaccess filtration units are in place leak tested after the initial installation and after any major modification or repair. In place leak testing after HEPA or

carbon replacement and the periodic in place leak tests for containment preaccess filtration units is not performed, as long as a) periodic visual inspections and pressure drop determinations across HEPA and charcoal filters are performed during scheduled shutdowns and b) charcoal adsorbers are sampled and laboratory tests are made to confirm performance at scheduled refueling shutdowns.

7. The containment purge and pressure relief system isolation valves' testing and inspection requirements are discussed in Section 6.2.4.

8. The pressure relief containment isolation valves are tested to determine the availability of the isolation function and the leak rate during reactor operation.

APPENDIX 9.4B - SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM

9.4B.1 DESIGN BASES

The heating and ventilation system is designed to maintain the ambient temperatures within the Service Water Intake Structure as indicated in Table 9.4-2 for normal operation. During emergency modes of operation, the temperature may rise to 132°F.

9.4B.2 SYSTEM DESCRIPTION

The Service Water Intake Structure heating and ventilating system consists of the following:

1. Service water pump area exhaust system

2. Intake structure heating system

3. Diesel-driven fire pump room exhaust system

The Service Water Intake Structure ventilation system is shown schematically on Figure 9.4-7.

The service water pump area exhaust system consists of eight 50-percent-capacity, propeller-type fans, which are mounted on a missile-protected exhaust plenum located in the overhead of the draft damper. The fans are started automatically by locally mounted temperature switches or manually by the intake structure operator.

A high temperature alarm in the Control Room is provided.

Outside air is drawn through grated openings with outside air intake screens in the floor of the intake structure located near the base of the pump motors. This air picks up heat dissipated by the service water pump motors. Then the stratified air is discharged by exhaust fans to the outdoors through a missile-protected opening in the side of the intake structure.

The diesel-driven fire pump room exhaust system consists of one 100- percent-capacity fan (rated at 8000 scfm), a gravity and fire damper, and associated exhaust ductwork. Air is exhausted from the diesel-driven fire pump room and adjacent electric fire pump area into the ventilation plenum located in the overhead of the intake structure. Outside air is drawn through grated openings in the floor of the intake structure by the exhaust system and passes over the equipment to pick up the heat generated by the fire protection components. This opening provides a means for the ventilation and diesel combustion air to enter the room and is also used to isolate the pump room in the event of a fire.

Six thermostatically controlled unit heaters, rated at 7.5 kW each, are provided to maintain the ambient temperature conditions within the Service Water Intake Structure to satisfy the winter design conditions. The primary reason for the use of unit heaters in the intake structure is to provide adequate freeze protection. The unit heaters are located around the perimeter of the intake structure and contain a sufficient degree of design margin to provide a reasonable working environment. The unit heaters are classified non-safety-related and are seismically supported and positioned so as not to damage any safety-related equipment housed within the intake structure.

The service water pump area exhaust system is not required to operate during the winter months. The diesel-driven fire pump exhaust system will operate only during the operation of the diesel engine to maintain the room free of any fumes. Intermittent operation of exhaust fan(s) to remove odors or purge the intake structure is left up to the discretion of the intake structure equipment operator(s).

9.4B.3 SAFETY EVALUATION

The reliability and safety of the Service Water Intake Structure ventilation system is ensured by the following features:

1. Redundancy in equipment and power supplies enables the system to sustain a single active component failure without loss of function.

2. Instrumentation and controls which incorporate audible and visual alarms in the Control Room facilitate continuous monitoring of system performances and alert the operator to system malfunctions.

3. The system is of seismic Category I and ANS Safety Class 3 design.

The Service Water Intake Structure is a seismic Category I structure; the HVAC equipment is designed to seismic Category I requirements and to withstand the tornado loads and tornado-related missile conditions described in Section 3.3.2.

9.4B.4 INSPECTION AND TESTING REQUIREMENTS

See Subsection 9.4C.1.1.

APPENDIX 9.4C - MISCELLANEOUS VENTILATION SYSTEMS

9.4C.1 DIESEL GENERATOR BUILDING VENTILATION SYSTEM

The Diesel Generator Building ventilation system has two functions: removal of excessive heat and provide heating during the winter months.

The design indoor conditions during normal plant conditions with Diesel Generator not in operation can be found in table 9.4-2. The temperature ranges during normal (while the Diesel Generators are being tested) and emergency conditions with Diesel Generator in operation are as follows:

Diesel Generator Rooms	40°F - 122°F
Day Tank Rooms	40°F - 122°F
Equipment Rooms	40°F - 130°F

Each diesel generator compartment is furnished with a ventilation system sufficient to remove heat from the area. The system consists of a set of intake and exhaust louvers and exhaust fans and is seismic Category I and ANS Safety Class 3. No redundancy is provided because the diesels are already redundant. Figure 9.4-4 shows the system schematically.

The Diesel Generator Building Ventilation System (DGBVS) consists of the following:

1. Diesel Generator Rooms Ventilation Sub-System (DGRVS)

2. Day Tank Rooms Ventilation Sub-System (DTRVS)

3. Heating System (Unit Heaters)

The DGRVS consists of (2) two independent trains, one (1) train for each Diesel Generator Area (DGA) (Train A and Train B). Each train consists of (4) four 25 percent capacity vaneaxial exhaust fans, dampers, louvers, ductwork and instrumentation necessary for the operation and control of the system.

The DTRVS consists of (2) two independent trains, one train for each DGA. Each train consists of one (1) 100 percent capacity centrifugal exhaust fan, dampers, louvers, ductwork and instrumentation necessary for the operation and control of the system.

Two (2) electric unit heaters are provided for each DGA (Train A and Train B). Instrumentation and controls are provided for each electric unit heater.

The diesel generator compartment ventilation system air exhausts are located in the roof of the respective compartments, and intake louvers are on the south and north walls of Unit 1 and Unit 2, respectively. The ventilation systems are seismic Category I and designed to withstand the tornado loads and tornado-related missiles described in Section 3.3.2.

The Unit Heaters are non safety but are required to meet the Seismic Category II requirements, so that in the event of an earthquake, the functioning of any Seismic Category I system or component is not adversely affected. The heaters are not required to be powered from class 1E power source because these are not required to operate during emergency condition.

Based on outside air temperature readings, selected diesel generator exhaust fans may be disabled to ensure design temperatures are maintained during winter months. In addition, certain intake louvers will be closed during the approximate time period the diesel generator exhaust fans are disabled.

To be certain that the ventilation fans are operating when the diesels are operating, the required number of fans in each diesel generator compartment start automatically on receipt of diesel generator start signal. During normal plant operation, the plant ventilation is operated manually.

Monitoring of the conditions in the diesel rooms is accomplished with the following Control Room instrumentation:

1. High diesel room temperature alarm

2. Ventilation fan motor tripped alarm

Since the diesel generator building ventilation (DGBV) fans are scheduled for annual maintenance, manual initiation of the diesel generator inoperable indication is provided, in accordance with R. G. 1.47.

Fans are powered from the same safety-related electrical bus as the diesel that is being ventilated.

The reliability and safety of the DGBVs is ensured by the following features:

1. Redundancy in equipment and power supplies enables the system to satisfy a single active component failure without loss of functions.

2. Instrumentation and controls which incorporate audible and visual alarms in the Control Room facilitate continuous monitoring of system performances and alert the operator to system malfunctions.

9.4C.1.1 Testing and Inspection

Shop inspection and testing are performed for equipment, including heating and controls.

The system is initially tested and adjusted for proper flow paths, flow capacities, heating and ventilating capacities, and mechanical operability.

Fan are rated and tested in accordance with the standards of the Air Moving and Conditioning Association (AMCA).

Heating coils are tested in accordance with manufacturer’s standards. Ductwork is tested in accordance with industry standards.

9.4C.2 MAIN STEAM AND FEEDWATER PIPING AREA VENTILATION SYSTEM

The main steam and feedwater piping area ventilation system is designed to provide adequate ventilation, to provide sufficient heat in order to ensure freeze protection of equipment during the winter months, and to remove the heat dissipated by the main steam and feedwater piping and any additional loads such as lighting or motor operators located in this area. This system is designed to maintain the ambient temperatures of the main steam and feedwater areas at between 40°F and 104°F during all normal modes of plant operation. This system is not required to function following a DBA or following a loss-of-offsite power. The Operator verifies the normal ventilation fans automatically trip following a safety injection signal and manually trips the fans if required.

Safety related equipment, located adjacent to the main steam and feedwater piping area are protected against the effects of main steam and feedwater pipe breaks by the use of isolation dampers located where the supply and exhaust ducts penetrate the main steam and feedwater piping area. These isolation dampers are Safety Class 3 and seismic Category I.

The main steam and feedwater piping area ventilation supply system consists of two 50-percent-capacity vane axial fans connected in parallel and located downstream of a supply housing. The supply housing consists of a roll-type roughing filter, cooling coil, and electric heating coil. The supply air is provided fromt he Electrical Area Ventilation system as described in Section 9.4C.3.

Gravity dampers are located downstream of each fan.

Air is provided to the main steam and feedwater ventilation supply system via a tempered air intake from the Electrical Area Ventilation System as described in Section 9.4C.3. The normal electrical area supply exceeds the capacity of the main steam and feedwater area system. Therefore, excess air is normally exhausted through the original fresh air intake. Because air that is drawn into the main steam and feedwater ventilation supply is tempered i.e., warmer in winter, cooler in summer than outside air, the performance of the main steam and feedwater ventilation system is enhanced. Outside air is drawn in, if required, by the supply fans through a louvered, roof-mounted penthouse. The air is filtered, tempered (heated or cooled), and then passed through the vane axial fans and supplied to the main steam and feedwater areas through a duct distribution system.

Each of the roll filters consists of a glass fiber media and casing with a motor operated media advance. The cooling coil is supplied with chilled water from the non-nuclear-safety-related plant ventilation chilled water system. The fans are direct-drive, vane axial type with totally enclosed air over motors. The vane axial fans are provided with adjustable pitch blades to facilitate balancing of the system. The electric heating coil located within the supply housing is the flange type.

The exhaust system consists of two 50-percent roof-mounted exhaust fans and associated ductwork. Air from the area is exhausted and passes through pneumatically actuated dampers, the exhaust fans, and gravity dampers, and is then discharged to atmosphere. The exhaust fans are the direct-drive, propeller type, complete with gravity dampers and a weatherproof housing. An in-duct radiation monitor is provided as described in Section 11.5.

The main steam and feedwater piping area ventilation system is shown schematically on Figure 9.4-4.

9.4C.3 ELECTRICAL AREA (SAFEGUARDS) VENTILATION SYSTEM

The electrical area (safeguards) ventilation system is designed to provide adequate ventilation in order to provide sufficient heat to ensure freeze protection of equipment during the winter months and to remove the heat dissipated by the electrical switchgear and associated electrical equipment (including lighting) contained within this area. The design indoor conditions during normal plant conditions can be found in Table 9.4-2.

The system is designed to maintain the ambient temperature between 40°F and 104°F during all normal modes of plant operation. This system is not required to function following a DBA. The Operator verifies the normal ventilation fans automatically trip following a safety injection signal and manually trips the fans if required. In addition, emergency fan coil units are provided to maintain the safeguards switchgear train A and train B areas below the maximum ambient temperature of 122°F in the event of loss-of-offsite power or a LOCA although the temperature may rise up to 129°F for a short period of time during a LOCA. The ductwork associated with the emergency fan coil units is interconnected with electrical area ventilation system ductwork. This ductwork, as well as the emergency fan coil unit, is classified Safety Class 3 in accordance with ANSI N18.2 and seismic Category I. The coils on the fan coil units are supplied by the safety chilled water system.

The tempered supply air is distributed via a common stairwell, open doors and through a ductwork system to all three electrical area elevations of the Safeguards Building. At elevations 810 ft 6 in. and 852 ft 6 in., emergency fan coil units are interconnected with the supply and exhaust ductwork and isolated from ductwork by Safety Class 3 gravity dampers. The emergency fan coil units are not normally required to operate. The electrical area ventilation exhaust system consists of two 50-percent capacity fans, gravity dampers, and associated ductwork. Air can be exhausted from each of the three electrical area elevations by the roof-mounted exhaust fans and discharged to the atmosphere. The normal exhaust for the Electrical Area Ventilation System is the Main Steam and Feedwater Area Ventilation Supply and its original fresh air intake as described in 9.4C.2, above. Because the system would not be balanced and excessive negative pressure could result during transients which stop the supply fans, the normal electrical area roof mounted ventilation exhaust may not be used when the Main Steam and Fedwater Area Ventilation System is running. One electrical area roof mounted exhaust fan may be run if only one Main Steam And Feedwater Area Ventilation supply fan is running and two Electrical Area Ventilation System supply fans are running.

The electrical area ventilation supply and exhaust system components are similar to the components as described in Subsection 9.4C.2, except for the rated air flow capacity.

The doorway between the electrical area and the equipment room is open to provide tempered air to the room to maintain minimum temperature and provide an exhaust path to the main steam and feedwater area ventilation supply as described above.

The electrical area ventilation system is shown schematically on Figure 9.4-4.

9.4C.4 CONTROL BUILDING UNCONTROLLED ACCESS AREA HEATING, VENTILATION, AND AIR-CONDITIONING SYSTEM

The uncontrolled access area HVAC system is designed to provide adequate ventilation, to provide sufficient heat in order to ensure freeze protection of equipment during the winter months and to remove the heat dissipated by equipment located in cable spreading rooms, battery rooms, and miscellaneous areas. The system maintains the ambient temperature between 40°F min. and 104°F max. during all normal modes of plant operation. It provides sufficient ventilation to the battery rooms to ensure a minimum number of air changes per hour in order to keep the hydrogen concentration of the rooms below the lower flammability limit, i.e., a hydrogen air mixture of 2.0 percent by volume. The battery room exhaust units and associated ductwork are designated ANS Safety Class 3 and seismic Category I and operate in the event of a DBA. Individual room heaters are supplied to maintain the battery rooms at 70°F min.

The supply system consists of two 50-percent-capacity vane axial fans connected in parallel and located downstream of a supply unit. The supply unit consists of a sheet metal housing which contains a roll filter, an electric heater, and a cooling coil which is supplied water from the non-nuclear-safety-related plant ventilation chilled water system. Outside air is drawn through louvered openings and an isolation damper, and passes through the supply unit, vane axial fans, and gravity dampers. The supply air is distributed throughout the Control Building and uncontrolled access areas, which contains cable spreading areas, battery and charger rooms, air compressor areas, and portions of a heating and ventilation equipment room.

The exhaust and recirculation system either returns air to the suction side of the supply unit or exhausts the air directly to atmosphere. This is accomplished through a system of exhaust ductwork located throughout the Control Building and uncontrolled access areas. The exhaust and recirculation system consists of two 50-percent-capacity vane axial fans and gravity dampers, both of which are required to direct the return airflow back to the supply unit or to the atmosphere.

A separate ventilation exhaust system is provided for each of the battery and charger rooms. Each battery room ventilation exhaust system consists of two 100-percent-capacity centrifugal fans, and each battery charger room ventilation exhaust system consists of two 100-percent-capacity axial fans; all the fans have pneumatically actuated isolation dampers located on the suction side of the fans and associated ductwork. Supply air is drawn into the battery and charger rooms from adjacent corridors by one of the two exhaust fans. The exhaust fans are arranged in parallel. The battery and charger room exhaust is discharged directly to the environs through a missile-protected roof exhaust vent.

If the supply air handling unit fails to deliver air due to loss of power or any other reason, make up air for the battery rooms is provided by dampers that open the HVAC equipment room to the atmosphere. The suction created by the battery room exhaust fans draws outside air via the HVAC equipment room and supply ductwork into the battery rooms.

Standby battery room and battery charger room exhaust fans are automatically actuated on receipt of an operating fan differential pressure trip signal. If, after a time delay, the standby fan fails to start, this condition is alarmed in the Control Room.

Electric unit heaters are located in each of the six battery rooms (three per unit). Each unit heater is an integral unit consisting of a propeller-type fan, fin tubular electric heating element,

casing, and thermostat. The battery room ventilation exhaust system is ANS Safety Class 3 and seismic Category I. The unit heaters in the safety related battery rooms are Class 1E.

The uncontrolled access area ventilation system is shown schematically on Figure 9.4-8.

9.4C.5 ELECTRICAL SWITCHGEAR AREA (TURBINE BUILDING) VENTILATION SYSTEM

The electrical switchgear area (Turbine Building) ventilation system requirements are to remove heat generated by the electrical switchgear equipment and to maintain the area ambient temperature between 104°F max. and 40°F min. during all normal mode of plant operation. This system is not required to operate in the event of a DBA.

The switchgear area ventilation requirement during summer operation is 30,000 scfm. The two 50-percent-capacity switchgear area vane axial supply fans, each rated at 15,000 scfm, are required to operate. The two 50-percent-capacity roof-mounted exhaust fans, each rated at 15,000 scfm, are also required during normal operating conditions. Winter operation requires only one vane axial supply fan (15,000 scfm) and one roof-mounted exhaust fan to provide heated ventilation air to the Turbine Building switchgear areas. The electric heating coil automatically operates, depending on the heating demand. The electrical switchgear area is shown schematically on Figure 9.4-3.

9.4C.6 OFFICE AND SERVICE AREA HVAC SYSTEM

The office and service area HVAC system is designed to provide a comfortable environment for operating personnel by maintaining the temperature as indicated in Table 9.4-2 for laboratories or other areas that contain calibration or test equipment. The office and service area HVAC system also prevents odors in the toilets, and locker rooms from diffusing to other occupied areas. Potentially radioactive rooms are exhausted through the plant atmospheric cleanup trains prior to releasing the effluents to the atmosphere. This system is designed to function during all normal modes of plant operation, but it is not required to operate in the event of a DBA or following a loss-of-offsite power. Running of the system affects the performance of the PPVS which exhausts air from a portion of areas served by this system. Redundant and diverse, single active failure proof trips of the Office and Service Area Fans are provided to ensure adequate ESF filtration.

The office and service area HVAC supply system consists of two 50-percent capacity air-handling units each supplying air through a duct distribution system to all the rooms of the office and service area. The air handling units are located in a heating and ventilation equipment room within the control building and consist of a gravity damper, electric heating coil, chilled water cooling coil, and centrifugal fan. Makeup air enters the system through a missile protected inlet. This makeup air is mixed with the return flow from the office and service area rooms and passes through roughing filter. The tempered air is then supplied to the rooms which constitute the office and service area.

The office and service area HVAC system also provides ventilation to the Alternate Access area for Unit 2 personnel to get into the Radiation Control area (RCA). This area is supplemented by the Primary Plant Ventilation system and two (2) fan coil units with auxiliary electric heating. These fan coil units utilize chilled water supplied by the Non-Safety Chilled Water system.

The Alternate Access area tool room is served by a split system air conditioner and electric duct heating coil to maintain the tool room at design condition. The tool room is maintained at a slightly negative pressure by exhausting 400 cfm through the Primary Plant ventiliation exhaust system.

The office and service area exhaust system consists of the following:

1. Toilet exhaust system

2. Filtered recirculation system

The office and service area toilet exhaust system consists of two 50-percent-capacity in-line exhaust fans, gravity dampers, and associated ductwork. The fans draw air from toilet areas, locker room, shower area, and janitor's closet and discharge it to the atmosphere.

The office and service area recirculation system consists of two 50-percent-capacity in-line fans, gravity dampers, and associated ductwork. The fans return air from all areas of the office and service area, except for the toilet and potentially radioactive areas. The fans discharge the return air back to the suction side of the air-conditioning units.

The office and service area is shown schematically on Figure 9.4-14.

9.4C.7 SECURITY OFFICE HVAC SYSTEM

The security office HVAC system is designed to provide a comfortable environment for the security personnel by maintaining the temperature between 68ºF and 80ºF during all normal modes of plant operation. The security office HVAC system also prevents odors in the toilet area from escaping to other occupied areas by exhausting this area directly to the atmosphere. The security office HVAC system is not required to operate in the event of a DBA or following a loss-of-offsite power. The security office HVAC system consists of an air-conditioning unit, roof-mounted toilet exhaust fan, and associated ductwork and accessories, see Figure 9.4-7.

9.4C.8 UNINTERRUPTABLE POWER SUPPLY (UPS) AND DISTRIBUTION ROOMS AIR CONDITIONING SYSTEMS

The UPS and Distribution Rooms Air Conditioning (A/C) System equipment is located at elevation 778'-0" of the Electrical and Control Building. The Fan Coil Units are located at elevation 792'-0" of the Electrical and Control Building. The system's primary function is to provide cooling for Class 1E electrical equipment in the UPS and Distribution Rooms, Trains A & B, Units 1 and 2, (elevation 792'-0") and the mechanical equipment room (elevation 778'-0"). The Fan Coil Units are designed to operate during all modes of plant operation. The air-conditioners are normally on stand-by. However, if a fan coil unit is unavailable, either one of the UPS air-conditioners (Train A or Train B), should be operating.

The Fan Coil Unit portion of the Air Conditioning System is comprised of one 100 percent capacity self-contained Fan Coil Unit. Each unit cools its respective UPS Room. The Heat Sink for the cooling coil is the Safety Chilled Water (SCW) System.

The other portion of the A/C system is comprised of two 100-percent capacity self-contained air-conditioning units located in adjacent rooms that are physically separated by a dividing fire

wall. The redundant A/C units are powered from independent Class 1E buses. The UPS and Distribution Room A/C system is ANS Safety Class 3 and Seismic Category I.

Each A/C unit consists of replaceable-type filters, refrigerant type (R-12) cooling coil, fan, compressor, condenser and all necessary instrumentation and controls. The heat sink on the condenser side of the units is supplied by the Component Cooling Water System (CCW).

The UPS and Distribution Room Air Conditioning system is shown schematically on Figure No. 9.4-15.

9.4C.9 BATTERY AND CHARGING ROOMS AIR CONDITIONING SYSTEM

The Battery and Charging Rooms Air Conditioning System equipment is located on elevation 830'-0" of the Turbine Building deck. The system's primary function is to provide cooling for the battery and charging rooms located on the same deck. The air-conditioning equipment is non-safety, non-seismic and will operate during normal plant operating conditions.

The air conditioning system is comprised of one vertical self-contained packaged unit. It is located in the charging room along with its thermostat. The air conditioning unit consists of replaceable type filters, refrigerant type R-22 cooling coil, heating coil, fan, compressor, condenser and condenser fan, and all necessary instrumentation and controls. The heat sink for the condenser side of the unit is supplied by outside air.

An exhaust fan with an explosion proof motor will provide a continuous exhaust for the battery room while the air conditioning system is operating.

If there is a malfunction of the air conditioning system and/or a level of hydrogen exceeding 2% is detected in the battery rooms, a local alarm will sound and the appropriate annunciator(s) will notify the operator in the control room.

The Battery and Charging Rooms Air Conditioning System is shown schematically on Figure No. 9.4-3.

9.4C.10 HIGH PRESSURE CHEMICAL FEED ROOM VENTILATION SYSTEM (HPCFVS)

The High Pressure Chemical Feed Room Ventilation System is designed to provide adequate ventilation to Room 100 to maintain the equipment below its design qualified temperature during Loss of Offsite Power (LOOP) and Loss of Coolant Accident (LOCA).

The HPCFVS consists of Class 1E power supply, safety related and seismically supported supply and exhaust fans. The supply fan is provided with two in-line gravity dampers upstream of the fan and ductwork to distribute outside air to the room. The exhaust fan is provided with two in-line gravity dampers downstream of the fan to exhaust air directly from the room to the outside.

The HPCFVS is required to operate when train "B" class 1E power source is utilized, and the temperature inside the room increases as a result of heat dissipated from the Motor Control Center (MCC) located in this room. The HPCFVS is to be started and stopped automatically by a temperature indicating switch (TIS) which senses and indicates the room temperature.

During all modes of plant operation, the HPCFVS will be in a standby mode. Room 100 will be maintained at a slightly negative pressure. However, when the HPCFVS is energized, the supply air to Room 100 is greater than the exhaust air to ensure the room is maintained at a positive pressure. The excess air will be exhausted through the primary plant exhaust system ductwork. The exhaust fan will trip if the supply fan fails to run.

INSPECTION AND TESTING REQUIREMENTS OF THE HPCFVS

SHOP TESTING:

Shop inspection and testing are performed for all equipment, including controls.

Each fan motor with fan blades attached, shall be statically and dynamically balanced prior to assembly.

After assembly, the fan shall be vibration tested at the fan rated speed. Results of the balancing test shall be certified.

FIELD TESTING:

The system is initially tested and adjusted for proper flow paths, flow capacities and mechanical operability to demonstrate field performance after the equipment has been installed.

Ductwork is tested in accordance with industry standards.

Fans are rated and tested in accordance with the standards of the Air Moving and Conditioning Association (AMCA).

Standby equipment is operated on a cyclic basis to ensure the availability of the equipment.

APPENDIX 9.4D - PLANT VENTILATION DISCHARGE VENT

9.4D.1 DESIGN BASES

The plant vent stacks are designed to aid in the dispersion of gaseous effluents exhausted by the primary plant ventilation system, including discharges of radioactive gas from the gas decay tanks (see Section 9.4.3 and 11.3), during normal operation.

9.4D.2 SYSTEM DESCRIPTION

The plant vent stacks are the release points of the primary plant ventilation system (see Figure 9.4-9). The stacks are located adjacent to the northwest side of the Unit 1 Containment and the southwest side of the Unit 2 Containment.

The design parameters for the plant vent stacks are the following:

1. Height of Release

   The height of the plant vents are approximately 196 ft above ground level. The release points are approximately 64 ft below the top of the Containment and at least 84 ft 6 in. above other adjacent structures.

2. Discharge Temperature

   As shown in Table 9.4-2, the minimum indoor design condition is 40°F. The temperature of the outside air is site related; therefore, the temperature difference between the gaseous effluent discharge and the outside air is also site related. The discharge temperature is dependent on ambient temperature conditions and mode of plant operation. The minimum discharge temperature is 40°F . The maximum is a function of the mode of plant operation. Information concerning outdoor design conditions is presented in Table 9.4-1. Design conditions (indoor) are tabulated in Table 9.4-2. See Figures 1.2-1 and 1.2-2.

3. Effluent Discharge Quantity

   The maximum quantity of gaseous effluent exhausted via each plant vent stack is shown on Figure 9.4-9.

   The exhaust flow from the ESF unit(s), during design basis events, will be discharged via the two vent stacks.

4. Plant Vent Stack Size and Shape

   The plant ventilation discharge vent is cylindrical in shape with a 66 inch diameter.

9.4D.3 SAFETY EVALUATION

The plant vents stacks are not designed as safety-related and no credit has been taken for the height of the stacks above plant grade in the offsite dose calculations. The vent stacks are

designed as seismic Category II. The system design is such that failure of the vent stack does not obstruct the release of gaseous effluent at the roof level and does not affect the functional integrity of the safety-related exhaust system.

APPENDIX 9.4E - PLANT VENTILATION CHILLED WATER SYSTEM

9.4E.1 DESIGN BASES

The plant ventilation chilled water system is designed to remove heat rejected by equipment and to maintain ambient temperature below design limits within the areas it serves.

This system provides chilled water to the following non-safety related cooling and ventilation systems of both units.

- Primary Plant Ventilation System

- Auxiliary Building HVAC Equipment Room System

- Safeguards Building Electrical Area Ventilation System

- Safeguards Building Main Steam and Feedwater Penetration HVAC System

- Electrical and Control Building Uncontrolled Access Area HVAC System

- Containment Air Recirculation System

- Containment Neutron Detector Well System

- Containment CRDM System

- Positive Displacement Charging Pump Fan Coil Unit

- Turbine Building Switchgear Area HVAC System

- Secondary Plant Sampling Package

- Turbine Building Office and Room Area Coolers

- Instrument Air System Coolers

The plant ventilation chilled water system is not required to mitigate the consequences of an accident or to maintain the plant in a safe shutdown condition. During a loss of offsite power, part of the chilled water system operates to provide chilled water to the containment cooling units and the positive displacement charging pump room fan coil units. It also provides chilled water to the secondary plant sampling package. All the other systems and equipment listed above are required only during normal plant conditions. The containment isolation valves are motor operated and are powered from the diesel generators.

The system is shown on Figure 9.4-11.

Chilled water is distributed by the recirculation pumps to the cooling coils of the systems listed above. The chillers are designed in accordance with the ASME B&PV Code, Section VIII.

The heat sink for the plant ventilation chilled water system are the Component Cooling Water System described in Section 9.2.2 and the Circulating Water System described in Section 10.4.5.

Except for the containment penetrations and containment isolation valves, the plant ventilation chilled water system is non-nuclear-safety and non-seismic. Therefore should a seismic event cause a loss-of-offsite power, the operation of the plant ventilation chilled water system would not be required. The Containment penetrations and the isolation valves are ANS Safety Class 2 and seismic Category I.

9.4E.2 SYSTEM DESCRIPTION

The plant ventilation chilled water system consists of two distinct subsystems as described below:

1. The first subsystem is shown on Figure 9.4-11 and consists of six chillers, four 50-percent capacity chilled water recirculating pumps, three 50-percent capacity chilled water booster pumps, a chilled water surge tank, chilled water cooling coils, and associated piping, valves and instrumentation. Four chillers are located in the Auxiliary Building and two chillers are located in the Unit 1 Turbine Building. The chillers located in the Auxiliary Building are of the centrifugal type with hermetically sealed, electric-motor-driven compressors. The chillers located in the Turbine Building are of the centrifugal type with open electric motor driven compressors. The pumps are electric-motor-driven, single stage, horizontally split, centrifugal type. The evaporators of the six chillers are connected in series. The four chilled water recirculating pumps are connected in parallel, as are the three chilled water booster pumps. This subsystem is operated to support simultaneous operation of, mainly, the containments of Unit No. 1 and Unit No. 2, which includes normal operation, reactor startup, normal reactor shutdown and reactor hot standby during a loss of offsite power.

   a. During reactor startup, normal operation, and normal reactor shutdown, operators will run a sufficient number of chillers to maintain ambient area temperatures within required limits.

   Additionally two out of four chilled water recirculating pumps and two out of three chilled water booster pumps are required to operate. The chilled water flow rate through the series Auxiliary Building chillers is 2400 GPM, and the chilled water flow rate through the series chillers in the Unit 1 Turbine Building is approximately 2600 GPM.

   The equipment served by this subsystem is as follows:

   - Three out of four Unit 1 Containment air recirculation and cooling units
   - One out of two Unit 1 Containment neutron detector well cooling units
   - Unit 1 Containment CRDM cooling unit
   - Three out of four Unit No. 2 Containment air recirculation and cooling units
   - One out of two Unit 2 Containment neutron detector well cooling units

- Unit 2 Containment CRDM cooling unit
- Unit 1 Positive Displacement Charging Pump Room fan coil unit
- Unit 2 Positive Displacement Charging Pump Room fan coil unit
- Secondary plant sampling package
- Unit 1 Turbine Building Switchgear Area supply unit
- Unit 2 Turbine Building Switchgear Area supply unit
- Turbine Building Office and Room Area fan coil units
- Instrument Air System Coolers

The component cooling water from the nonsafeguard loops of the CCWS for both Units is used as the cooling medium for the four chillers located in the Auxiliary Building. The arrangement of the CCWS piping allows the condensers in a specific pair of these chillers to be connected in series. Two chillers are cooled by Unit 1 non-safeguard Component Cooling Water loop; and the other two chillers are cooled by Unit 2 non-safeguard Component Cooling Water loop. The Component Cooling Water flow rate is 2000 GPM per two series condensers in a pair, or a total of 4000 GPM for the three operating chillers.

The two chiller condensers in the Unit 1 Turbine Building are connected in parallel and are supplied with circulating water as the cooling medium. Each condenser receives 2000 GPM of circulating water i.e., a total of 4000 GPM for the two operating chillers.

b. During a loss of offsite power, this subsystem is required to serve Unit 1 and Unit 2 containment cooling units and Unit 1 and Unit 2 positive displacement charging pump room fan coil units. It will also serve the secondary plant sampling package. Only the four chillers located in the Auxiliary Building and two associated chilled water recirculating pumps are required to operate. The two Unit 1 Turbine Building chillers and their associated chilled water booster pumps are isolated. The chilled water flow rate through the four operating series chillers is 2400 GPM. The CCW flow rate through each operating pair of series chillers is 2000 GPM i.e., a total of 4000 GPM for both operating pairs. The four chillers and the two chilled water recirculating pumps which are selected for operation are powered from the Standby Power System during a loss of offsite power, as described in Section 8.3.

Remote manually actuated air operated valves (Tag nos: X-HV-6056A and X-HV-6057A) are utilized to switch this system from the normal to post-loss of offsite power operation, by isolating the Unit 1 Turbine Building Chillers and associated chilled water booster pumps and vice-versa.

2. The second plant ventilation chilled water subsystem is shown on Figure 9.4-11 and consists of three chillers, each sized for 33.33 percent of the total cooling capacity, four 33.33-percent capacity chilled water circulating pumps, a chilled water surge tank

(which is shared with the first subsystem), chilled water cooling coils, and associated piping, valves and instrumentation. The chillers are centrifugal type, with direct open, electric-motor driven compressors. The chilled water circulating pumps are electric-motor-driven, single stage, horizontally split centrifugal type. The evaporators of the chillers are connected in parallel to common chilled water supply and return headers. The four chilled water circulating pumps are also connected in parallel. This subsystem is operated during normal operation, reactor startup and normal reactor shutdown only. It does not operate during a loss of offsite power. During operation at maximum design load, all three chillers are required to operate to generate a combined cooling capacity of 1980 tons. Additionally, three out of four chilled water circulating pumps are required to operate, with the fourth pump remaining on standby. The chilled water flow rate through the three parallel chillers is 650 GPM per chiller i.e., a total flow rate of 1950 GPM.

The equipment served by this subsystem is as follows:

- Six out of eight Primary Plant Ventilation System (PPVS) supply units
- The Auxiliary Building HVAC Equipment Room Supply Unit
- Unit 1 Safeguards Building Electrical Area supply unit
- Unit 1 Safeguards Building Main Steam and Feedwater Penetration Area supply unit.
- Unit 2 Safeguards Building Electrical Area supply unit
- Unit 2 Safeguards Building Main Steam and Feedwater Penetration Area supply unit.
- Units 1 and 2 Electrical and Control Building Uncontrolled Access Area supply unit.
- Instrument Air System Coolers

The condensers of the three chillers are connected in parallel and are supplied with circulating water as the cooling medium. Each condenser receives 2250 gpm of circulating water i.e., a total of 6750 gpm.

The surge tank is provided to accommodate expansion and contraction within the system and to permit monitoring of the system for leakage.

The makeup water to the tank is supplied from the Demineralized Water System.

Except for cooling units located inside the Containment, Instrument Air System and those for the Turbine Building Office and Room Area, all cooling coils are provided with a thermostatically controlled valve at the outlet of the coil. The thermostatically controlled valve for the Turbine Building Switchgear Area cooling coils is located at the inlet of the coils. The outlet air temperature from the coil controls the valve.

The coils for the cooling units located inside the Containment, except for the CRDM cooling coils, are provided with an electric-motor-operated-on-off valve which is interlocked with the respective cooling unit fan. Flow through the CRDM coils is not affected when the fans, which can be controlled from the Control Room, are stopped.

Vent and drain connections are provided on piping and equipment where necessary to facilitate testing and maintenance operations. All major components are provided with upstream and downstream isolation valves to facilitate maintenance operation.

The system is not required to function following a DBA. During a DBA, the chilled water supply and return Containment penetrations isolation valves are closed, and the system will shut down.

The chilled water is circulated through the system by the chilled water recirculating pumps and circulating pumps.

Although the two plant ventilation chilled water subsystems described above, operate independently of each other, they are interconnected with piping and with proper valving so as to permit, selectively and through administrative decision, the switching of chilled water supply to various cooling equipment, from the first subsystem to the second subsystem, under off-normal conditions (e.g., failure of a chiller, pump, a cooling unit, etc.).

The operating pumps take suction from their respective chilled water return headers and discharge it to the evaporator sections of the associated operating chillers. The cooled chilled water, after rejecting heat to the refrigerant in the chillers, is discharged into the respective chilled water supply main. The chilled water from the supply main flows through the cooling coils of the various cooling units associated with the subsystems. The warm chilled water from the cooling coils flows via the chilled water piping back to the respective chilled water return header, which feeds the suction of the chilled water recirculating pumps, thus completing the closed system.

Instrumentation and controls incorporating audible and visual annuciation facilitate continuous monitoring of the system performance and alert the operator in the event of a system malfunction. To maintain constant chilled water flow, a valve located in a recirculation loop is modulated by a signal from a flow element in the first subsystem and a pressure differential controller in the second subsystem. The Containment isolation valves are motor operated and are powered from the diesel generators.

9.4E.3 SAFETY EVALUATION

The reliability of the plant ventilation chilled water system is ensured by the following features:

The Containment penetrations and isolation valves are ANS Safety Class 2 and seismic Category I.

APPENDIX 9.4F - SAFETY CHILLED WATER SYSTEM

9.4F.1 DESIGN BASES

The safety chilled water system is designed to remove heat rejected by ESF pump motors, Uninterruptable Power Supply (UPS) Equipment and electrical switchgear and to maintain ambient temperature below design limits within the rooms it serves. The chilled water from the system is supplied to the safety-related cooling units. The system is also used during normal operations to aid in heat removal. The system is shown on Figure 9.4-12.

The heat sink for the safety chilled water system is the safety-related Component Cooling Water System described in Section 9.2.2.

The safety chilled water system is ANS Safety Class 3 and seismic Category I. The system is required to operate during all modes of operation.

The chillers are designed in accordance with ASME B&PV Code Sections III & VIII. The piping is designed in accordance with ASME B&PV Code Section III.

Each unit is provided with two 100-percent-capacity chilled water systems each of which are powered from independent Class 1E buses. Thus, the system will sustain a single active component failure without loss of function during all modes of plant operation.

9.4F.2 SYSTEM DESCRIPTION

The safety chilled water system is shown on Figure 9.4-12.

The system for each unit consists of two-100-percent-capacity hermetic centrifugal chillers, two-100-percent-capacity chilled water recirculation pumps, a chilled water surge tank, chilled water fan coil units, and associated piping, valves, and instrumentation. Each chiller is rated for 101 tons of refrigeration at design conditions. Each pump is capable of circulating chilled water at 300 gpm design flow. The required component cooling water flow is 300 gpm at design conditions.

The chilled water by the system is supplied to the cooling units as follows:

1. Component cooling water pump emergency fan coil units

2. Charging pump emergency fan coil unit

3. Spent fuel pool heat exchanger and pump emergency fan coil unit (These fan coil units are common for both units, and the piping arrangement will allow chilled water to be supplied from either the Unit 1 or the Unit 2 chilled water system.)

4. Safety injection pump emergency fan coil unit

5. Containment spray pump emergency fan coil unit

6. RHR emergency fan coil unit

7. Auxiliary Feedwater pump emergency fan coil unit

8. Electric area emergency fan coil units

9. Uninterruptable Power Supply (UPS) Room Fan Coil Units

The chillers are of the centrifugal type with hermetically sealed, electric-motor-driven compressors. The component cooling water from the respective trains of the safeguards loop of the CCWS is used as a cooling media for the condensers.

The pumps are electrical-driven, single-stage, horizontal-split, and centrifugal type.

The surge tank is provided to accommodate expansion and contraction within the system and to permit monitoring of the system for leakage. The makeup water to the tank is supplied from either the Demineralized Water System or the Reactor Makeup Water System, with consideration given to water chemistry for the Safety Chilled Water System corrosion inhibitor. The partition in the surge tank provides separate surge volumes for each safety train. A leak in one train will not affect the other train.

Vent and drain connections are provided on piping and equipment where necessary to facilitate testing and maintenance operations. All major components are provided with upstream and downstream isolation valves to facilitate maintenance operation.

The system is required to operate during post DBA conditions. The system is powered from Class 1E safety buses. During loss-of-offsite power, the power is provided from diesel generators.

Chilled water is circulated through each of the two closed-loop safety trains. The recirculation pump takes suction from the chilled water return line and the chilled water surge tank which connects into the return line. The recirculation pump discharges into the evaporator of the chiller. The then chilled water from the evaporator enters into the supply header and passes through the fan coil units connected in parallel. The return chilled water from the fan coil units enters into the chilled water line, thus completing the closed system.

Except for the electrical area and Uninterruptable Power Supply (UPS) Room fan coil units, all fan coil units are interlocked with respective equipment to start. The fan coil units for the electrical area start on a Safety Injection or Blackout Sequence Signal. The chilled water recirculation pumps start on a Safety Injection, Blackout Sequence Signal or the start of a CCW Pump.

The instrumentation and controls for the safety chilled water system are provided for automatic and remote operation of the system. The operation and supervision of the safety chilled water system following a Control Room evacuation can be accomplished by using the system components local controls (for fan coils), the Remote Shutdown Panel, and the Switch Transfer Panel for Train "A" pump and chiller.

9.4F.3 SAFETY EVALUATION

The safety chilled water system is ANS Safety Class 3 and seismic Category I.

The reliability of the chilled water system is ensured by the following features:

1. The use of two 100-percent-capacity systems, one for each train, and the power supply from redundant Class 1E buses to each electrically operated equipment; therefore, the system meets the requirement of a single failure without loss of function.

2. Instrumentation and controls are available, providing audible and visual annunciation. This permits continuous monitoring of system performance.

3. The safety chilled water system components are located in a seismic Category I structure.

9.4F.4 INSPECTION AND TESTING REQUIREMENTS

See Subsection 9.4.2.4.

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 FIRE PROTECTION SYSTEM

9.5.1.1 General

This section is a description of the Fire Protection Program of the CPNPP units 1 and 2. The evaluation of fire hazards is included in the CPNPP Fire Protection Report (FPR) which follows the format of the U.S. Nuclear Regulatory Commissions's "Supplementary Guidance on Information Needed for Fire Protection Program Evaluation" and the supplementary criteria in their September 30, 1976, letter.

The overall Fire Protection Program was developed utilizing the defense in depth concept. This concept is a combination of:

1. Preventing fires from starting

2. Quickly detecting and suppressing fires that do occur to limit the extent of damage

3. Designing plant safety systems so that a fire that becomes fully established and burns for a considerable time, in spite of the fire protection systems provided, will not prevent essential plant safety functions from being performed.

The FPR quantifies potential fire hazards throughout the plant in terms of combustible heat release loading. The Fire Protection and Detection Systems are designed based on this heat release loading and on the nature of the transient and in situ combustible material in the area. A summary of this information is presented in tabular form in the FPR.

9.5.1.2 Method of Analysis

9.5.1.2.1 Definitions

Several terms with their definitions as they relate to the Fire Protection Program for CPNPP are presented below. Unless the terms are noted below, the definitions are as stated in Section I of Branch Technical Position APCSB 9.5-1 Reference [2].

1. Fire area

   The fire area is that section of a building or the plant that is separated from other areas of the plant by fire barriers with openings and penetrations protected by seals or closures having a fire resistance rating equal to the rating assigned to the barrier. The fire areas extend through more than one elevation where plant design requirements and low amounts of combustible material in a specific area allow. These areas are designated on FPR Figures.

2. Fire Barriers

   Fire barriers are those components of construction (walls, floors, or protective coverings) that are rated by approved laboratories or are constructed in accordance with the

requirements stated by authorities having jurisdiction in hours of resistance to fire and used to prevent spread of fire.

3. Fire Zone

The fire zone is a subdivision or portion of a fire area that is designated on the FPR Figures.

4. Fire Duration

Fire duration is the approximate time expressed in minutes that the tabulated combustible material will burn. The duration is based on the heat release that will produce an exposure equivalent to the standard time-temperature curve (ASTM E-119).

5. Fire Break

The fire break is a physical barrier that prevents fire propagation, that is, the spreading of a fire from one component to another or the direct exposure of a component to the heat and flames of a fire, or both.

6. Design Basis Fire

Design basis fire is a fire that is postulated to occur in a fire area or fire zone assuming no manual, automatic, or other firefighting action has been initiated. The combustibles in the area are totally consumed and the fire burns at a rate modeling the standard time-temperature curve (ASTM E-119).

7. Enclosed

The term “enclosed” means being surrounded by a case which will prevent a person from accidentally contacting live electrical parts. It also applies to flammable liquids which are contained or encased in fire-resistant materials or buildings and to barriers which may or may not be fire rated that surround or encompass fire areas or fire zones.

8. Dry

The term “dry” indicates that the connecting piping between a deluge valve and the nozzles of a water system is not normally pressurized with water.

9. Wet

The term “wet” indicates that the connecting piping between the main loop and a hose station isolation valve or water nozzle is normally pressurized with water.

10. Radiation Zone

Radiation zone is the classification of an area based on the expected dose equivalent rate (mrem/hr) within that area. See Section 12.3 for a detailed description.

11. Fire Safe Shutdown Essential System or Component

    An essential system or component is defined as a system or component which is required to be operational to safely shutdown the plant in the event of a fire.

12. Maximum Permissible Fire Loading

    The Maximum Permissible Fire Loading (MPFL) is the maximum fire loading (BTU/sq ft) which can be expected to be contained within a fire area by the fire area boundaries without compromising safe shutdown capability.

13. Fire Hazards Analysis Evaluation

    A Fire Hazard Analysis Evaluation is an assessment of the impact of a single fire hazard on redundant components or systems used to provide fire safe shutdown functions for the plant. A Fire Hazards Analysis Evaluation is performed by a Fire Protection Engineer and, if required a Systems Engineer. The purpose of a Fire Hazards Analysis Evaluation is to demonstrate compliance with BTP APCSB 9.5-1 Appendix A based on the following considerations:

    - potential transient and in situ combustible hazards are considered.

    - protection provided is commensurate with the hazards.

    - the consequences of a fire on the plant's ability to safely shutdown are considered.

    - The Fire Hazards Analysis Evaluation is written, organized and maintained to facilitate review by a person who is not involved in the evaluation.

    - The conclusions of the FHA Evaluations are summarized in the applicable sections of the Fire Protection Report.

9.5.1.2.2 Assumptions

The FHA Evaluation is based on the following assumptions:

1. Generally, the minimum fire barrier rating is three hours except for the barriers enclosing the stairwells and elevator shafts, which are rated at two hours, the cable tray/conduit fire barriers which are rated at 1-hour, one hour fire rated cable, and other special cases where a rating of less than three hours is adequate.

2. When it is determined that a fire involving a fire safe shutdown component or system will not affect its redundant counterpart, the redundant system is assumed to operate without failures.

3. The Maximum Permissible Fire Loading for a fire zone assumes a fire burning in the area which follows the characteristics of the standard time-temperature curve, or as noted in the FPR, Reference [19].

4. A fire involving a combustible loading, up to the Maximum Permissible Fire Loading for the fire zone, will be contained within the fire area by the passive and active/fire protection features (i.e. fire wall and sprinklers, etc.). Furthermore, it is assumed that if any of these passive or active fire protection features is inoperable and the compensatory actions required by the Fire Protection Report have been implemented then an equivalent level of protection is provided.

9.5.1.2.3 Methodology

In order to evaluate potential fire hazards, provide adequate fire protection, ensure isolation of fire safe shutdown systems from these hazards, and prevent the release of radioactive material to the environment, the following method of design and analysis has been formulated and implemented for the entire plant:

1. The plant is divided into separate fire areas using plant walls and floors as barriers. Due consideration as-shown below is given to the separation of redundant fire safe shutdown components from each other, from non-fire safe shutdown components and from major concentrations of combustible materials. Considerations were also given to other area characteristics such as electrical cable routing into and through the area, the ductwork supplying and exhausting the area, access and egress routes for the area, and vent area for depressurization during a tornado.

2. For each fire area/fire zone, the heat of combustion for each in-situ combustible is calculated. The calculated heat of combustion for all in-situ combustibles is divided by the floor area to determine the combustible loading (BTU/sq ft) for the fire area/fire zone. In addition, the approximate fire duration (minutes) is determined based on the ASTM E-119 standard time-temperature curve, (except as noted in the FPR Section II). The combination of transient combustibles and the in-situ combustibles will not exceed the Maximum Permissible Fire Loading without implementation of compensatory measures.

3. The fire safe shutdown essential equipment in each area is tabulated.

4. Once the fire area and combustible material information is tabulated, fire protection equipment is located throughout the plant based on the severity and configuration of the fire hazards, the calculated heat release of each fire area and the plant equipment and components located in the fire area.

5. Fire detectors are located in all areas of the plant where there is a significant combustible loading and in all areas containing equipment required for safe shutdown except as described in Section 9.5.1.6.1.

6. Hose stations are installed in all safety related buildings of the plant such that an effective hose stream can reach any location in a safety related building except as described in Section 9.5.1.6.1.

7. Portable extinguishers are located in all safety related buildings in accordance with NFPA 10 requirements.

8. Fixed automatic water suppression systems will generally be installed in safety related plant areas where any of the following conditions exist:

   a. A high fire hazard exists

   b. Redundant safe shutdown equipment or cabling outside the Containment Building is located in the same fire area and is not separated by a three hour fire barrier.

   c. There is a congestion of cabling.

   In areas where condition (a) and in areas where condition (b) described above exists, the type of protection that will be provided as a minimum will be a sprinkler system providing coverage adequate for the hazard in the area unless justification for deviations are provided per reference [19] and as described in 9.5.1.6.1. The water spray design density will be based on Section 9.5.1.6.1-E.3.c.

   Where the condition described in (c) exists, based on Section 9.5.1.D.3.c, sprinkler systems will be provided for cabling to augment other fire protection features in the area.

9. Where redundant fire safe shutdown equipment cabling is located in the same fire area and is not separated by a three hour fire barrier or a horizontal distance of 20 feet with negligible intervening combustibles or fire hazard, one train of this cabling, if not one hour fire rated cable, will be enclosed by a one-hour fire barrier (or radiant energy shield inside containment) unless an alternate shutdown path is utilized or justification for deviations are provided per reference [19] except as described in Section 9.5.1.6.1.

10. The Cable Spreading Room contains equipment and cables belonging to both safety trains. The following fire protection systems will be provided:

    a. Hose stations for manual fire fighting

    b. Fixed Halon primary suppression system

    c. Manual pre-action sprinkler system

    d. Automatic fire detection system

    e. An alternate shutdown system

11. The plant will be capable of being safely shutdown in the event any of the fires postulated in the Fire Protection Report occurs. Alternate shutdown systems and procedures have been developed using shutdown paths available to the operator which are either free from fire damage or otherwise controllable in spite of such fire damage.

9.5.1.3 Fire Hazard Analysis Evaluation

See Reference [19], Fire Protection Report

See Reference [50] regarding Independent Spent Fuel Storage Installation (ISFSI) Haul Path and ISFSI Pad

9.5.1.4 Fire Protection System Description

9.5.1.4.1 General

The Fire Protection System detects, alarms, and extinguishes fires. It is comprised of two subsystems: Fire Detection and Fire Suppression.

The Fire Detection System is a plant-wide system designed to detect fires in the plant, alert the Control Room operators, and alert the plant fire brigade of the fire and its location.

The Fire-Suppression System is designed to extinguish any fire postulated to occur in the Fire Protection Report. It is comprised of a water supply system, fixed water sprinkler and spray systems, Halon systems, fire hose stations, and portable extinguishers.

9.5.1.4.2 System Design Parameters

1. Fire Detection System

   The Fire Detection System consists of the following components:

   a. Fire Detectors

      1. Ionization smoke detectors

      2. Thermal heat detectors

      3. Ultraviolet detectors

      4. Thermistor line detectors

   b. Fire Detection Local Control Panels

      These panels provide local indication of the status of the protected area. Indication provided is annunication of alarms and system trouble status. These panels also provide automatic initiation of fire suppression where applicable.

   c. Fire Detection Main Control Panel

      This panel is located in the control room. Any fire alarm that is detected in the plant will alarm on this panel in the control room. Trouble circuits of each local panel are also monitored.

The fire detectors are strategically located throughout the plant to detect, annunciate, and indicate in the Control Room, the location of a fire.

The power supplies for the Plant Fire Detection system meets the requirements of NFPA 72D Section 2220. The Plant Fire Detection system can be provided power from any one of the following eight (8) sources: two (2) main generators, two (2) offsite power supplies, and four (4) standby diesel generators (plant emergency power supply).

The fire detection system is electrically supervised for a wiring break in the detection and alarm circuits. Loss of supervision causes an audible and visual trouble indication on the main fire detection control panel, located in the Control Room, in accordance with NFPA 72D requirements. Thermistor-line fire detection systems are supervised for a break or short circuit of the sensing element. Ground fault supervision is provided except as noted in Section 9.5.1.6.1-E.1.

Ionization detectors are of the two-chamber-type design. The first chamber is a reference chamber to compensate for sensitivity changes caused by temperature, barometric pressure, and humidity variations. The second chamber is a sensing chamber open to the outside elements through a protective screen which permits combustion products to enter, while preventing insects and foreign matter from entering and causing false alarms.

Thermal detectors are of the fixed-temperature, rate compensation types or continuous strip thermistor line type.

Ultraviolet detectors respond directly to the presence of flame by sensing the ultraviolet radiation emanating from the flame.

2. Fire Suppression Systems

a. Water Supply Systems

The water supply system was designed using NFPA Codes and BTP 9.5-1 Appendix A. The water supply network and the arrangement of the water extinguishing systems are shown on Figures 9.5-43 through 9.5-48 and Figures 9.5-61 and 9.5-62. The water extinguishing systems are designed to operate with the shortest portion of the Fire Protection yard-loop out of service. The water storage capacity is based on supplying water to the largest fixed extinguishing system and the manual hose stream requirements of Appendix A.

Three 50 percent pumps (one electric motor-driven, two diesel engine-driven; each rated at 2000 gpm at a Total Dynamic Head of 370 ft) are provided for protection of both units. In addition, a jockey pump (rated at 60 gpm at a Total Dynamic Head of 300 ft) maintains the required water pressure throughout the system at all times.

The fire pumps are located in a separate fire pump house structure and take suction from two dedicated fire water storage tanks. The fire pumps are single stage horizontal splitcase centrifugal pumps and Underwriters' Laboratory, Inc. listed for fire protection service.

A 12-in. diameter piping network encircles the entire plant. Post indicator gate valves isolate sections of the yard fire-loop and each branch connection extending off the loop. These valves have indicators that show valve position.

Hydrants isolated by individual auxiliary gate valves (curb box valves), are provided adjacent to the yard loop at approximately 250 ft intervals. Each hydrant is equipped with at least two 2-1/2 in. hose outlets. A 2-1/2 in. outside independent hose gate valve is connected to each outlet to provide individual

control of the hoses. A hose house is provided adjacent to each hydrant. Each 2-1/2 in. hose house is equipped with the following accessories:

1. One fire axe
2. One crowbar
3. Two hydrant wrenches
4. Six coupling spanners
5. Two hose and ladder straps
6. Two 2-1/2 in. hose washers (spares)
7. One emergency light
8. Three hundred feet of 2-1/2 in. firehose
9. Two adjustable straight stream-fog type nozzle and play pipe combination

All hose connections and nozzles are provided with NH type threaded connections that are compatible with those of the local fire department.

b. Fixed Water Suppression Systems

Automatic and manual suppression systems typically consist of a hydraulically balanced piping network, an isolation valve supervised as described in Section 9.5.1.6.1-E.3.6, fusible-link, quartzoid bulb or directional solid-cone spray nozzles, a water flow alarm, and a local and remote (control room) audible and visual alarm. The systems are designed and balanced to provide water spray densities as described in Section 9.5.1.6.1-E.3.c. The fixed water suppression systems are divided into six categories. Each category is as follows:

1. Automatic wet-pipe sprinkler systems

   Automatic wet-pipe sprinkler systems are provided for general area coverage and to protect electrical cable trays.

2. Automatic Water Spray Systems

   When a fire is detected its location is annunciated on the local fire detection control panel and in the control room. The Fire Detection Local Control Panel, transmits a signal to open the proper deluge valve. A water-flow alarm sounds locally and in the control room indicating water flowing through the piping network. The deluge valves operate automatically as described above, or can be manually operated locally. Once actuated, deluge valves can only be reset manually.

Water spray systems are provided for the following:

- Turbine Building Hazards (i.e. Lube Oil, Feedwater Turbines, Hydrogen Seal Oil)

Water spray systems are also provided for the following atmospheric cleanup units' (ACUs) charcoal adsorber beds:

- Four Containment preaccess units

Each charcoal adsorber bed is equipped with thermistor strip type heat detectors.

Upon detection of an abnormally high temperature in any one of the charcoal adsorber beds, a high temperature signal is generated locally and in the control room. This signal also initiates demineralized water flow to the respective unit. Should demineralized water be unavailable, the operator is alerted in the control room and he can manually route fire protection water through the demineralized water pipe supplying these water spray systems.

If the temperature of the charcoal adsorber bed continues to rise, the detection system will generate a high-high temperature signal which will automatically open the deluge type valve on the ACU and initiate water spray on its charcoal bed. When the temperature of the charcoal adsorber bed drops below the high-high set point, the deluge valve will close automatically. Deluge type valves for ACUs can only be operated automatically.

The water flow requirements for each unit are based on the manufacturer's suggested flow rate to extinguish a charcoal-adsorber fire.

3. Manual Water Spray Systems

Manually operated water spray systems protect each turbine generator main shaft bearing at locations where the main shaft is exposed to the environment. These systems are provided with an approved deluge valve, a piping network, a flow switch and directional solid-cone spray nozzles. Isolation valves are provided for each exposed section of the shaft to preclude spraying unaffected areas. Manual operation as well as annunciation of a fire alarm condition via the plant fire detection system is provided.

Water spray systems are also provided for the following atmospheric cleanup units' charcoal adsorber beds:

(a) Sixteen primary plant exhaust ventilation units.

(b) Two hydrogen purge exhaust units.

(c) Two Control Room emergency filtration units.

(d) Two Control Room emergency pressurization units.

Each charcoal adsorber bed is equipped with thermistor strip type heat detectors.

Upon detection of an abnormally high temperature in any one of the charcoal adsorber beds, a high temperature signal is generated locally and in the control room. The fire suppression system headers to the ACUs are normally dry. Manual operation of a deluge charging valve charges the headers with water. If the temperature of the charcoal bed continues to rise, the detection system will generate a high-high temperature signal which will automatically open the deluge type valve on the ACU and initiate water spray on its charcoal bed. When the temperature of the charcoal bed drops below the high-high set point, the deluge valve will close automatically. Deluge type valves on the units can only be operated automatically.

4. Manual Preaction Sprinkler Systems

A manual preaction sprinkler system consists of a normally dry piping network, automatic sprinkler heads, latching normally-closed deluge valve, water flow alarm (which transmits an alarm to the Control Room via the local fire detection panel indicating water flow in the piping), and an approved isolation valve complete with tamper switch.

Manual preaction sprinkler systems are provided in areas requiring general area and electrical cable tray protection but where the potential for inadvertent water discharge is to be eliminated.

Plant areas, protected by manual preaction sprinkler systems, include the following:

- Cable Spread Room systems are provided for backup suppression capability for the Halon Systems.

- Switchgear and Hot Shutdown Panel room systems are provided for specific hazards.

Upon detection of fire, the deluge valve is manually opened through operation of an electric pull station allowing water flow into the piping network. Water will only discharge into the protected area through the individual automatic sprinkler heads which witness their rated temperature.

5. Manual Deluge Sprinkler System

A manual deluge sprinkler system protects the Fuel Building loading bay and consists of a normally dry piping network, open sprinkler heads,

deluge valve, water flow alarm and an approved isolation valve with tamper switch.

Upon detection of a fire, the deluge valve is manually operated through operation of an electric pull station allowing water flow into the piping network which will discharge through all sprinkler heads protecting the area.

6. Automatic Preaction Sprinkler Systems

Automatic preaction sprinkler systems protect the diesel fuel oil day tank rooms and consist of a normally dry piping network, automatic sprinkler heads, approved deluge valve, water flow alarm (which transmits an alarm to the control room via the local fire detection panel indicating water flow in the piping), and an approved isolation valve complete with tamper switch.

When a fire is detected by the detection system, the local fire detection control panel transmits a signal to open the system deluge valve. Water will only discharge into the protected area through the individual automatic sprinkler heads which witness their rated temperature. The deluge valves operate automatically as described above, or can be manually operated locally. Once actuated, deluge valves can only be reset manually.

c. Halon Extinguishing Systems

Automatic, total flooding, fixed, Halon extinguishing systems, actuated by ionization detectors, are provided for the cable spreading rooms. Manual actuated, total flooding, Halon extinguishing systems, are provided for the plant computer rooms.

Halon concentrations for each area are in accordance with NFPA suggested concentration except as described in Section 9.5.1.6.1. Each system is provided with two charges of Halon.

d. Fire Hose Stations

Fire hose stations are located strategically throughout the plant for manual fire fighting operations.

The fire hose stations located on the operating deck of the Turbine Building are equipped with 100 ft of 2-1/2 in. hose using NFPA 14 as a guideline for Class 1 service except as noted in Reference [19].

All other fire hose stations throughout the plant are equipped with 100 ft of 1-1/2 in. hose and a nozzle compatible with the type of fire hazard in the area. The respective standpipes which supply water to these hose stations are sized and located throughout the plant using NFPA 14 for class II service as a guideline except as noted in 9.5.1.6.1 and Reference [19].

Fire hose stations are also provided to supply makeup water to the spent fuel pools if normal sources are unavailable. Fire hose stations (EL. 860', see Figure 9.5-47 Sheet 1) in the area of the spent fuel pools are capable of supplying makeup in excess of the boil-off rate. Since the fire protection makeup is from the SSI, the supply of water is essentially unlimited.

The Hose stations for the Containment Building's Control Room and the cable spreading rooms are dry pipe with manual charging required.

The hose stations inside the containment are fed from the Demineralized Water System via a transfer pump. This pump is initiated by hand pull stations inside the containment. Should demineralized water not be available or the system malfunctions, additional hand pull stations are provided inside the containment which allows for the normal fire protection water to be used for fire fighting purposes. The use of demineralized water will minimize the possiblity of introduction of chlorides inside the containment.

e. Portable Fire Extinguishers

Portable fire extinguishers are provided for fire suppression throughout the plant. The quantity and type of extinguishers located in each fire area are based on the type, quantity, and specific hazard conditions in the respective fire areas. All extinguishers are in accordance with the guidelines of NFPA pamphlet No. 10.

9.5.1.4.3 System Description

Fixed water suppression systems, fixed Halon systems, hose stations, and portable extinguishers are used as the primary and secondary means of fire suppression. As shown on Figures 9.5-43, 9.5-61 and 9.5-62 water is supplied to the water suppression systems and the hose station standpipes from two atmospheric storage tanks via an underground piping distribution system and three 50 percent-capacity fire pumps, one electric motor driven and two diesel engine driven.

When pressure in the main loop drops below a set pressure point, the jockey pump starts automatically; when the system is repressurized, the jockey pump stops automatically. If the jockey pump cannot maintain the system pressure, the lead fire pump (the electric motor-driven fire pump) automatically starts. The diesel engine-driven fire pumps sequentially start automatically if the system pressure cannot be maintained by the other pumps for a preset period of time. The three fire pumps can only be shut down manually at the fire pump house location. The three fire pumps are connected to an approved flow meter to facilitate periodic testing of any fire pump. Pressure switches for the three pumps are located between the pump check valve and pump isolation valve to prevent starting an isolated pump. A siamese fire department connection is provided for emergency fill of the system by a fire truck or a portable auxiliary pump. This fill is used as a backup to the pumps. As required by NFPA No. 24, a check valve and a ball drip valve are provided at the connection of the siamese to the main loop. The siamese connection is located adjacent to the Service Water Intake Structure.

As shown in Figures 9.5-44 and 9.5-44A, the Turbine Buildings have an internal loop which supplies the standpipes. This internal loop has multiple connections to the underground loop in Unit 1 and Unit 2. Crosstie lines are provided between the Unit 1 and Unit 2 Turbine Buildings to

facilitate isolation of sections of either loop. Valves are provided in accordance with NFPA 14 to isolate the system. The water spray systems and automatic sprinkler systems are connected to the outside loop via isolation valves located in the fire protection valve rooms in the basement of each Turbine Building. The valve rooms are accessible from inside and outside the Turbine Buildings, as required, to control the water flow to the suppression systems.

As shown on Figure 9.5-46 Sheet 2, the water suppression systems protecting the diesel generators are independently supplied from the main yard loop. Actuation of the wet-pipe sprinkler system protecting one of the diesel generators will not affect the operation of the other diesel generator. Each diesel generator compartment is provided with a watertight door to prevent flooding of the adjacent areas.

The fire pump house structure is divided into five rooms with three hour rated fire barriers. The structure is protected by an automatic wet-pipe sprinkler system. Water flow and valve tamper alarms are provided at the pump house location and in the Control Room. Each room in the fire pump house is provided with detection which annunciates locally and in the Control Room.

The automatic wet pipe, manual preaction, automatic preaction, manual deluge, hose stand pipe and water spray systems are supplied by the respective safety related building interior supply loop.

Yard post indicator valves are located in supply lines to permanent plant auxiliary buildings in accordance with NFPA 24 to shut off the water supply to these buildings.

The Fire Protection System piping that is located throughout the Auxiliary, Electrical and Controls, and the Safeguards Buildings is routed via a sectionalized loop located on the basement elevation: 778'-0" in the Electrical and Controls Building and 790'-6" in the Auxiliary and Safeguards Buildings. The loop which is located in the corridor remote from the safeguards equipment is supplied via two lines running through the instrument and service air compressor area and a third line running through the corridor adjacent to the laundry holdup area. The loop and supply lines are normally wet (i.e., pressurized with water) and the branch connections off the loop to the hose stations and water spray systems are normally dry and only become pressurized during a fire condition. The Fire Protection System piping in these buildings is designed in accordance with the requirements of Reference [43] to assure the system pressure boundary integrity.

Based on the system design requirements and the physical layout of the Fire Protection System supply lines and main loop, the worst assumed crack in the wet piping would result in spraying a motor control center, which is designed per NEMA requirements and provided with drip shields and thus not effect cold shutdown of the plant. Also, the loop is provided with sectionalizing valves as noted previously in this section to isolate portions in event of a crack. A crack is not considered in the dry portion of the System.

Each Cable Spreading Room Halon system consists of a detection system, storage cylinders, manifold and header assembly, control valves, piping, nozzles, and local control panels. The Halon is released automatically after receipt of a fire signal from cross zoned ionization detectors located in the respective area. Each system incorporates a time delay which provides a warning for personnel evacuation of the area. The reserve charge of Halon is provided for automatic protection of the areas during the time the main cylinders are being refilled following a discharge. Halon can also be released manually.

Each Unit Computer Room Halon system is manually actuated by control room personnel and each system consists of storage cylinders, manifold and header assembly, control valves, piping, and nozzles. An ionization detection system provides control room notification of a fire.

A Fire Detection System is provided throughout the plant. When a fire is detected, it is annunciated and indicated by zone on the fire detection panel in the Control Room. An alarm-indicating lamp illuminates the base of the ionization detector showing the actuated detector. The majority of detectors are placed overhead in the monitored areas. Detectors serve a dual purpose: 1) they sound an alarm via the Control Room main fire detection panel and 2) where applicable, they actuate the automatic suppression systems.

9.5.1.4.4 Administrative Controls

The administrative controls related to fire protection at CPNPP are contained in the CPNPP Fire Protection Report and augmented by station procedures as required for effective implementation. These include the limiting conditions for operation required for the fire suppression systems and fire detection instrumentation, the corresponding compensatory measures, and the required surveillance test requirements to assure operability of the systems which were previously contained in the technical specifications. The licensee may make changes to the approved fire protection program without prior approval of the commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

9.5.1.5 Plant Fire Protection Design Requirements

9.5.1.5.1 General Plant Arrangement

The various buildings of CPNPP are divided into a series of fire areas. The primary consideration in this division was the separation of fire safe shutdown systems and components from their redundant counterparts and the isolation and separation of fire hazards from fire safe shutdown systems. Consideration was also given to the isolating of combustibles not located in, or exposed to, areas containing fire safe shutdown components and to provide access and egress routes to fire areas for plant personnel and the fire brigade.

9.5.1.5.2 Structural Construction Elements

All structural construction elements are composed of noncombustible materials. Structural walls, floors, and ceilings consist of poured, reinforced concrete, concrete block or structural steel framing with precast concrete panels or metal siding. Where these assemblies are designated as fire barriers, (i.e., 1 hour, 2 hour or 3 hour fire ratings) the construction is in accordance with a Underwriters' Laboratory listed design, a Uniform Building Code design, a specific fire test by a nationally recognized laboratory or as described in Section 9.5.1.6.1.

9.5.1.5.3 Interior Construction Elements

1. Building Partition Characteristics

   Interior walls and partitions which are non-load bearing and designated as fire barriers are constructed of reinforced concrete, concrete block or noncombustible steel studs and gypsum dry wall. Interior non-load bearing walls which act as fire barriers are constructed in accordance with a Underwriters' Laboratory listed design, a uniform building code

design, a specific fire test by a nationally recognized laboratory, or as described in Section 9.5.1.6.1.

Radiation shielding is accomplished using reinforced concrete poured in place, removable concrete block barriers, lead wool, lead brick, leaded glass and high density silicone materials. The lead wool, lead glass and lead brick are not used when the barrier is designated as both shielding and fire barrier. The thickness of the concrete varies as required to meet the specified shielding, but in all cases is greater than the required thickness of a three-hour rated fire barrier.

Where wood construction is used in the Turbine Building all of the wood used is treated with flame retardant.

Interior finishes such as gypsum plaster, ceramic tile, and acoustical ceiling materials are noncombustible. The acoustical tiles are mineral fiber board with a flame spread rating of less than 25 in accordance with ASTM E-84, Surface Burning Characteristics of Building Materials. Containment building protective coating systems have a flame spread rating in accordance with ANSI N101.2-1972. Fire retardant paints and coatings conform to Factory Mutual requirements. When practical, all other non-fire retardant paints and coatings have flame spread and smoke density ratings less than or equal to 25 when tested in accordance with ASTM-84. Vinyl asbestos floor tiles, located in the Control Room, various corridors, and in the office areas in the Turbine Building, have a flame propagation index of less than four. The flame propagation index is in accordance with UL 992, Test Method for Measuring the Flame Propagating Characteristics of Flooring and Floor Covering Materials. Carpeting installed in the Control Room is discussed in Section 9.5.1.6.2.

Steel checker plate hatch covers and removeable concrete hatches are provided in floor openings required for equipment removal. In floors designated as fire barriers, the steel checker plate hatch covers are coated with an approved fire-resistant coating. Protection provided by steel hatch covers has been demonstrated through analysis, in lieu of providing a tested configuration, as described in 9.5.1.6.2. Concrete hatches are constructed such that the designated fire rating of the barrier is maintained.

The reflective piping insulation is composed of stainless steel sheets and foil and the thermal piping insulation is composed of hydrous calcium silicate. Both are 100-percent inorganic and will not burn or support combustion. Anti sweat piping insulation is composed of fiberglass and along with its finishing cement has been tested by UL to the requirements of ASTM E-84 with a flame spread of 25, a fuel contribution of 25, and a smoke development of 50.

2. Penetration Seals

Penetrations in designated fire barriers are sealed with an approved fire stop material except as noted in Subsection 9.5.1.6.2. The penetration seals have fire resistance ratings that meet or exceed the rating designated for the barrier. The majority of the penetrations are sealed with approved silicone materials tested in accordance with the requirements of ASTM E 119 and, in the case of electrical seals, tested in accordance with IEEE 634.

3. Fire Door Assemblies

Door openings in designated fire barriers are provided with approved labeled fire door assemblies except as noted in Subsection 9.5.1.6.2.

9.5.1.5.4 Ventilation System Characteristics

1. Fire Dampers

All ductwork that penetrates a designated fire barrier of two hours or greater is equipped with an approved damper with a rating at least equivalent to that designated for the barrier. Most fire dampers are equipped with heat-responsive elements which automatically release the fire damper blade when the air temperature in the ductwork exceeds the predetermined element operating temperature. Where appropriate, fire dampers are equipped with electro-thermal links. Fire dampers are normally open, but they close during a fire condition. Where applicable, fire dampers located in ductwork are seismically qualified to ensure that the dampers will not close during a seismic event (see Section 9.4.5).

2. Smoke Removal

Smoke will be removed by portable smoke ejectors. Directions for approaching the fire area, placing of the smoke ejectors and routing of the ejector trunk for each fire area will be provided to the fire brigade. This information will be based on the smoke removal study.

The radiation level of the smoke in the subject fire area will be measured. If it is within the allowable limits as outlined in 10 CFR-20, smoke ejection can proceed. However, if the radiation is above the acceptable limits one of the below procedures must be followed:

a. The radiation level will be allowed to decay until it is below the acceptable limit, the smoke then will be released, directly to the atmosphere.

b. The radioactive smoke will be passed through the atmospheric clean-up units, before being released to the atmosphere.

The primary method of smoke removal is via the use of electrically powered portable smoke ejectors. In addition, a sufficient number of gasoline engine powered portable smoke ejectors are provided as a back-up.

Smoke control and heat venting in the Turbine Building will be accomplished by power venting of the mezzanine level using the exhaust fans. Venting of the basement level will be accomplished by natural draft through grating and equipment hatches and openings in the mezzanine floor.

9.5.1.5.5 Electrical Cable and Cable Tray Design - Characteristics

Generally, electrical cables are flame-retardent and nonpropagating in nature and conform to the criterion of IEEE 383- 1974. They will not support combustion in the absence of a sustained ignition source. The cable construction will allow wetting down without structural damage or

electrical faulting. All cable trays, conduits, and their supports are constructed of noncombustible materials.

Outside the Containment buildings, where cable trays containing cabling related to both redundant trains of equipment required to bring the plant to a hot standby condition, and where both trains are located in the same fire area, and are not separated by a negligible combustible horizontal distance of greater than or equal to 20 feet, and are not comprised of one hour fire rated cable, one train of cabling will be protected by at least a one hour rated fire barrier. Where this situation exists, automatic sprinklers are arranged to provide coverage adequate for the hazards in the area. Sprinklers are also provided for cabling where there is a congestion of cable trays see Section 9.5.1.6.1d. Fire stops are provided within the cable trays whenever the cables penetrate walls or floors designated as fire barriers. Fire stops are not provided at intermediate points in vertical or horizontal cable runs, except in long vertical runs. In such instances, fire stops are located at intervals equivalent to floor spacings. It is a general installation practice that vertical tray runs are provided with solid, sheet steel covers for a minimum distance of 4 feet above the floor where necessary for physical protection of the cable. Fire stops are not provided in cable trays inside the Containment Buildings. Conduit fire stops are provided when the conduit penetrates a designated fire barrier and is not run continuously through the fire area (as described in Section 9.5.1.6.1.D.3.d).

9.5.1.5.6 Transformers

All interior transformers are of the air-cooled dry type and do not contain any insulating oil. The main, unit auxiliary, station service and startup transformers are oil-cooled and are located outdoors adjacent to the Turbine Buildings. The main transformers are separated from each other, as well as from the Turbine Buildings, by a blank three-hr rated fire wall. The unit auxiliary transformer and startup transformers XST1 and XST2 are separated from the Turbine Buildings by a three-hr rated fire wall. Penetrations in this wall within 50 ft from each side of the center line of the transformer are protected in order to maintain the fire-resistant integrity of the wall. Additional walls are provided extending out from the Turbine Building wall to protect the ventilation openings located in the exterior Turbine Building wall. Station service transformers 1ST and 2ST and alternate transformer XST2A are separated from the Turbine Building by a distance greater than 50 feet. Alternate startup transformer XST1A is separated from adjacent structures by a three-hour rated fire wall.

9.5.1.5.7 Flammable Liquid and Gas Storage

1. Flammable Liquid Storage

   All significant amounts of flammable liquids are stored in separate fire areas that are isolated from the adjacent plant areas by three-hr fire rated barriers. As a minimum, fire detectors are provided in each area, and dependent upon the hazard, a fixed fire extinguishment system is provided. In all instances, such areas do not present a potential hazard for equipment located in the adjacent areas.

2. Flammable Gas Storage

   Bulk storage of all flammable explosive gases is located outside the primary, secondary and turbine plant buildings. The storage facility is an open structure located outdoors in

the yard adjacent to the security fence. An explosion or fire in this area will not affect any of the primary plant buildings.

9.5.1.6 Conclusions

9.5.1.6.1 Comparison with Appendix A of Branch Technical Position APCSB 9.5-1 of Standard Review Plan 9.5.1

As requested by the NRC in their September 30, 1976, letter, the following is a comparison of the CPNPP fire protection program with the guidelines in Appendix A to the above branch technical position.

CPNPP Fire Protection Program

The CPNPP Fire Protection Program is established to ensure that a fire will not prevent safe plant shutdown and will not endanger the health and safety of the public. Fire protection at CPNPP is accomplished using a defense-in-depth approach to include fire detection and extinguishing systems and equipment, administrative controls and procedures, and trained personnel. CPNPP is committed to meeting the requirements contained in the Fire Protection Report.

APCSB 9.5-1 Appendix A

A. Overall Requirements of the Fire Protection Program

A.1 Personnel

Responsibility for the overall fire protection program should be assigned to a designated person in the upper level of management. This person should retain ultimate responsibility even though formulation and assurance of program implementation is delegated. Such delegation of authority should be to staff personnel prepared by training and experience in fire protection and nuclear plant safety to provide a balanced approach in directing the fire protection programs for nuclear power plants.

CPNPP Fire Protection Program

The CPNPP Fire Protection Program meets the requirements of the D. B. Vassallo letter of August 1977 entitled “Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance,” as described in Section 13.3B.

APCSB 9.5-1 Appendix A

A.1 The fire protection staff should be responsible for:

(a) coordination of building layout and systems design with fire area requirements, including consideration of potential hazards associated with postulated design basis fires,

(b) design and maintenance of fire detection, suppression, and extinguishing systems,

(c) fire prevention activities,

(d) training and manual fire-fighting activities of plant personnel and the fire brigade.

CPNPP Fire Protection Program

See Section 13.3B.1.

APCSB 9.5-1 Appendix A

A.1 Subsequently, the FSAR should discuss the training and the updating provisions such as fire drills provided for maintaining the competence of the station fire-fighting and operating crew, including personnel responsible for maintaining and inspecting the fire protection equipment.

CPNPP Fire Protection Program

See Section 13.3B.1.

APCSB 9.5-1 Appendix A

A.1 The qualification requirements for the fire protection engineer or consultant who will assist in the design and selection of equipment, inspect and test the completed physical aspects of the system, develop the fire protection program, and assist in the fire-fighting training for the operating plant should be stated.

CPNPP Fire Protection Program

The personnel who are responsible for the design and selection of equipment, inspection and testing of the completed physical aspects of the system, and development of the fire protection program are Members of the Society of Fire Protection Engineers or equivalent and are knowledgeable in fire protection and detection systems design as well as in the requirements of nuclear plant safety.

APCSB 9.5-1 Appendix A

A.2 Design Bases

The overall fire protection program should be based upon evaluation of potential fire hazards throughout the plant and the effect of postulated design basis fires relative to maintaining ability to perform safety shutdown functions and minimize radioactive releases to the environment.

CPNPP Fire Protection Program

The overall Fire Protection Program is based on the evaluation of potential fire hazards. See Subsections 9.5.1.2.2 and 9.5.1.2.3 for further discussion.

APCSB 9.5-1 Appendix A

A.3 Backup

Total reliance should not be placed on a single automatic fire suppression system. Appropriate backup fire suppression capability should be provided.

CPNPP Fire Protection Program

Appropriate secondary fire suppression capability is provided throughout the plant by fixed water suppression systems, fire hose stations, or portable extinguishers.

APCSB 9.5-1 Appendix A

A.4 Single Failure Criterion

A single failure in the fire suppression system should not impair both the primary and backup fire suppression capability. For example, redundant fire water pumps with independent power supplies and controls should be provided. Postulated fires or fire protection system failures need not be considered concurrent with other plant accidents or the most severe natural phenomena.

CPNPP Fire Protection Program

The fixed fire suppression systems are designed so that a single failure cannot impair both it and the secondary fire suppression system for the particular area. System design parameters and descriptions are stated in Subsection 9.5.1.4.

APCSB 9.5-1 Appendix A

A.4 The effects of lightning strikes should be included in the overall plant fire protection program.

CPNPP Fire Protection Plan

CPNPP is provided with a plantwide lightning protection system. Fire suppression capability is provided such that any fire started by lightning which would affect the plant can be extinguished.

APCSB 9.5-1 Appendix A

A.5 Fire Suppression Systems

Failure or inadvertent operation of the fire suppression system should not incapacitate safety related systems or components. Fire suppression systems that are pressurized during normal plant operation should meet the guidelines specified in APCSB Branch Technical Position 3-1, “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.”

CPNPP Fire Protection Program

Failure or inadvertent operation of the fire suppression system will not incapacitate both paths of systems or components required for safe shutdown. Redundant trains or components that are susceptible to damage from water spray are either physically separated or protection is provided from the effects of water spray. Fire suppression systems located in the Safeguards, Electrical and Control, Auxiliary and Fuel Handling buildings that are pressurized during normal plant operation are in accordance with the specified guidelines. See FSAR Section 3.6B.

APCSB 9.5-1 Appendix A

A.6 Fuel Storage Areas

The fire protection program (plans, personnel and equipment) for buildings storing new reactor fuel and for adjacent fire zones which could affect the fuel storage zone should be fully operational before fuel is received at the site.

CPNPP Fire Protection Program

The Fire Protection Program for the Fuel Building is in effect and the suppression systems operational.

APCSB 9.5-1 Appendix A

A.7 Fuel Loading

The fire protection program for an entire reactor unit should be fully operational prior to initial fuel loading in that reactor unit.

CPNPP Fire Protection Program

The Fire Protection Program for Unit 1 will be in effect and the suppression systems operational prior to initial fuel loading. The same constraint applies to Unit 2.

APCSB 9.5-1 Appendix A

A.8 Multiple Reactor Sites

On multiple-reactor sites where there are operating reactors and construction of remaining units is being completed, the fire protection program should provide continuing evaluation and include additional fire barriers, fire protection capability, and administrative controls necessary to protect the operating units from construction fire hazards. The superintendent of the operating plant should have the lead responsibility for site fire protection.

CPNPP Fire Protection Program

Separation of Unit 1 from Unit 2 is accomplished by the use of equivalent three hour rated fire walls in areas where fire walls are required and by security barriers where fire walls are not required. The Manager, Plant Operations has the lead responsibility for site fire protection.

APCSB 9.5-1 Appendix A

A.9 Simultaneous Fires

Simultaneous fires in more than one reactor need not be postulated, where separation requirements are met. A fire involving more than one reactor unit need not be postulated except for facilities shared between units.

CPNPP Fire Protection Program

This criterion has been incorporated in evaluating the Fire Protection Program for CPNPP

APCSB 9.5-1 Appendix A

B. Administrative Procedures, Controls and Fire Brigade

B.1 Administrative procedures consistent with the need for maintaining the performance of the fire protection system and personnel in nuclear power plants should be provided.

CPNPP Fire Protection Program

Administrative procedures and controls will be utilized to ensure the reliable performance of the Fire Protection System and personnel.

APCSB 9.5-1 Appendix A

B.2 Effective administrative measures should be implemented to prohibit bulk storage of combustible materials inside or adjacent to safety-related buildings or systems during operation or maintenance periods.

CPNPP Fire Protection Program

Effective administrative controls will be utilized to minimize the amount of combustibles that may be exposed in safety-related areas during operation or maintenance period. These controls will govern the following:

1. Proper storage and handling of flammable gases and liquids, HEPA and charcoal filters, dry, unused ion exchange resins and other combustible supplies.

2. Transient fire loads during maintenance and modifications such as flammable liquids, woods, plastic equipment and other combustibles. This control will require an in-plant review of work activities to identify transient fire loads. The supervisor or foreman responsible for supervising the work activity will specify any required additional fire suppression.

3. Waste, debris, scrap, and oil spills resulting from a work activity in a safety-related area will be minimized while work is in progress and removed upon completion of the activity.

4. Periodic inspection for accumulation of combustibles.

APCSB 9.5-1 Appendix A

B.3 Normal and abnormal conditions or other anticipated operations such as modifications (e.g., breaking fire stops, impairment of fire detection and suppression systems) and refueling activities should be reviewed by appropriate levels of management and appropriate special actions and procedures such as fire watches or temporary fire barriers implemented to assure adequate fire protection and reactor safety. In particular:

CPNPP Fire Protection Program

Effective administrative controls will be utilized to protect safety-related equipment from fire damage or loss resulting from work involving ignition sources such as welding, cutting, grinding, or open flame work. Administrative controls will prohibit the use of open flame or combustion smoke for leak testing and other ignition sources in certain areas. Administrative controls will be established to ensure that the following pre-cautions are taken:

APCSB 9.5-1 Appendix A

B.3 (a) Work involving ignition sources such as welding and flame cutting should be done under closely controlled conditions. Procedures governing such work should be reviewed and approved by persons trained and experienced in fire protection. Persons performing and directly assisting in such work should be trained and equipped to prevent and combat fires. If this is not possible, a person qualified in fire protection should directly monitor the work and function as a fire watch.

CPNPP Fire Protection Program

1. Welding, cutting, grinding, or open flame work will be authorized by the responsible foreman or supervisor through a work permit. The responsible foreman or supervisor will have received sufficient firefighting and fire prevention instruction on the use of fire

extinguishers and methods to combat Class A, B and C fires. A fire watch will be established if required by the work permit.

APCSB 9.5-1 Appendix A

B.3 (b) Leak testing, and similar procedures such as air flow determination, should use one of the commercially available aerosol techniques. Open flames or combustion generated smoke should not be permitted.

CPNPP Fire Protection Program

2. Open flames or combustion generated smoke will not be used for leak testing.

APCSB 9.5-1 Appendix A

B.3 (c) Use of combustion material, e.g., HEPA and charcoal filters, dry ion exchange resins or other combustible supplies, in safety related areas should be controlled. Use of wood inside buildings containing safety-related systems or equipment should be permitted only when suitable non-combustible substitutes are not available. If wood must be used, only fire retardent treated wood (scaffolding, lay down blocks) should be permitted. Such materials should be allowed into safety related areas only when they are to be used immediately. Their possible and probable use should be considered in the fire hazard analysis to determine the adequacy of the installed fire protection systems.

CPNPP Fire Protection Program

3. The CPNPP Fire Protection Program uses administrative procedures to control combustibles within safety related areas of the plant.

APCSB 9.5-1 Appendix A

B.4 Nuclear power plants are frequently located in remote areas, at some distance from public fire departments. Also, first response fire departments are often volunteer. Public fire department response should be considered in the overall fire protection program. However, the plant should be designed to be self-sufficient with respect to fire fighting activities and rely on the public response only for supplemental or backup capability.

CPNPP Fire Protection Program

The CPNPP Fire Protection Program is designed for plant self-sufficiency for both fire fighting activities and equipment. Public fire response is regarded as supplemental capability. The public fire department will be called to the site after assessment of the fire condition by the responsible

fire brigade personnel. Fire protection procedures and equipment are compatible with those of the public fire department

APCSB 9.5-1 Appendix A

B.5 through B.7

CPNPP Fire Protection Program

See Section 13.4

APCSB 9.5-1 Appendix A

C. Quality Assurance Program

Quality assurance (QA) programs of applicants and contractors should be developed and implemented to assure that the requirements for design, procurement, installation, and testing and administrative controls for the fire protection program for safety related areas as defined in this Branch Position are satisfied. The program should be under the management control of the QA organization. The QA program criteria that apply to the fire protection program should include the following:

CPNPP Fire Protection Program

Luminant Power has developed a comprehensive Quality Assurance program (QA program) as described in Section 17.0. Luminant Power has delegated responsibility for the coordination of design, construction and operation of CPNPP to the Nuclear Generation organization. The structures, systems and components covered by the Quality Assurance program are listed in Table 17A-1. The provisions of the QA program apply to all activities which affect the safety-related functions of those systems and components. The preparation, review, and approval of Fire Protection Program procedures will be handled in the same manner as other safety-related procedures.

APCSB 9.5-1 Appendix A

C.1 Design Control and Procurement Document Control

Measures should be established to assure that all design-related guidelines of the Branch Technical Position are included in design and procurement documents and that deviations therefrom are controlled.

CPNPP Fire Protection Program

The fire protection system designer and/or supplier will assure, through a program of design reviews and reviews of procurement documents that the guidelines of the Branch Technical Position are reflected in these documents. Deviation from the requirements of these documents

will be controlled consistent with the original requirements. The applicant shall verify the adequacy of these activities by audit and surveillance.

APCSB 9.5-1 Appendix A

C.2 Instructions, Procedures and Drawings

Inspections, tests, administrative controls, fire drills and training that govern the fire protection program should be prescribed by documented instructions, procedures or drawings and should be accomplished in accordance with these documents.

CPNPP Fire Protection Program

The fire protection program will provide for the availability and use of documented instructions, procedures, or drawings to perform inspections, tests, administrative controls, fire drills and personnel training related to the system. The availability and use of these will be verified through audit and surveillance.

APCSB 9.5-1 Appendix A

C.3 Control of Purchased Material, Equipment and Services

Measures should be established to assure that purchased material, equipment and services conform to the procurement documents.

CPNPP Fire Protection Program

Measures will be established to assure that materials and services related to the fire protection service conform to the procurement documents. These measures will provide for a program of receiving inspection and supplemented by subsequent tests. Procurement activities, whether performed by the applicant or a Fire Protection System supplier, will be covered by the applicant's audit and surveillance program or Underwriters Laboratory surveillance programs.

APCSB 9.5-1 Appendix A

C.4 Inspection

A program for independent inspection of activities affecting fire protection should be established and executed by, or for, the organization performing the activity to verify conformance with documented installation drawings and test procedures for accomplishing the activities.

CPNPP Fire Protection Program

The applicant will, through a program of audit or surveillance, assure that involved organizations, performing activities related to fire protection, establish and execute an internal program of independent inspection. The independent inspection function will verify conformance.

APCSB 9.5-1 Appendix A

C.5 Test and Test Control

A test program should be established and implemented to assure that testing is performed and verified by inspection and audit to demonstrate conformance with design and system readiness requirements. The tests should be performed in accordance with written test procedures; test results should be properly evaluated and acted on.

CPNPP Fire Protection Program

The applicant will verify by a program of audits and surveillance that required inspections are defined, that the tests demonstrate by appropriate acceptance criteria, conformance with design and system readiness requirements. The applicant will verify that tests are performed in accordance with written test procedures, that tests results are properly evaluated, and that observed deficiencies are corrected.

APCSB 9.5-1 Appendix A

C.6 Inspection, Test and Operating Status

Measures should be established to provide for the identification of items that have satisfactorily passed required tests and inspections.

CPNPP Fire Protection Program

The applicant will verify through, audit and surveillance, that tests and inspections are documented as required by specified documentation requirements to provide adequate identification of items that have satisfactorily passed those required tests and inspections.

APCSB 9.5-1 Appendix A

C.7 Non-Conforming Items

Measures should be established to control items that do not conform to specified requirements to prevent inadvertent use or installation.

CPNPP Fire Protection Program

Those items that do not conform to specified requirements will be identified and controlled until proper disposition is made to prevent inadvertent use or installation.

APCSB 9.5-1 Appendix A

C.8 Corrective Action

Measures should be established to assure that conditions adverse to fire protection, such as failures, malfunctions, deficiencies, deviations, defective components, uncontrolled combustible material and non-conformances are promptly identified, reported and corrected.

CPNPP Fire Protection Program

The applicant will assure through a program of testing and inspection that conditions adverse to fire protections, such as failures, malfunctions, deficiencies, deviations, defective components, uncontrolled combustible material and nonconformance are properly identified, reported and corrected.

APCSB 9.5-1 Appendix A

C.9 Records

Records should be prepared and maintained to furnish evidence that the criteria enumerated above are being met for activities affecting the fire protection program.

CPNPP Fire Protection Program

The applicant will assure, through procurement and programmatic requirements, as well as audit and surveillance, that adequate documentation will be prepared and maintained to serve as evidence that the fire protection program is in conformance with the above requirements.

APCSB 9.5-1 Appendix A

C.10 Audits

Audits should be conducted and documented to verify compliance with the fire protection program including design and procurement documents; instructions; procedures and drawings; and inspection and test activities.

CPNPP Fire Protection Program

Audits will be conducted and documented to serve as evidence that the applicant has assured that activities including design, procurement, instructions, procedures, inspections and tests are in compliance with the fire protection program.

APCSB 9.5-1 Appendix A

D. General Guideline for Plant Protection

D.1 Building Design

D.1.a Plant Layouts should be arranged to:

(1) Isolate safety-related systems from unacceptable hazards, and

CPNPP Fire Protection Program

(1) All buildings of the plant are divided into fire areas. The criteria used to develop this arrangement are discussed in Subsection 9.5.1.2.2, 9.5.1.2.3 and 9.5.1.5.1.

APCSB 9.5-1 Appendix A

(2) Separate redundant safety related systems from each other so that both are not subject to damage from a single fire hazard.

(2) Alternatives:

(a) Redundant safety-related systems that are subject to damage from a single fire hazard should be protected by a combination of fire retardant coatings and fire detection and suppression systems, or

(b) a separate system to perform the safety function should be provided.

CPNPP Fire Protection Program

(2) (a): Where redundant fire safe shutdown systems, required to bring the plant to a hot standby condition, are located within the same fire area and are subject to damage from a single fire hazard a Fire Hazards Analysis Evaluation demonstrates and documents compliance to that recommended in the guideline by protecting the function with one of the following:

For systems located both inside and outside the Containment Building, see FPR Section II (4.5), Reference [19].

(b) Where a redundant system required to bring the plant to a cold shutdown condition is subject to damage from a single fire hazard, the following will be provided:

1) Fire detection system

2) procedure to repair at least one train of the damaged system within 72 hours.

APCSB 9.5-1 Appendix A

D.1.b In order to accomplish 1.(a) above, safety related systems and fire hazards should be identified throughout the plant. Therefore, a detailed fire hazards analysis should be made. The fire hazards analysis should be reviewed and updated as necessary.

CPNPP Fire Protection Program

The CPNPP Fire Protection Program is based on detailed fire hazard evaluations which satisfy this guideline.

APCSB 9.5-1 Appendix A

D.1.c For multiple reactor sites, cable spreading rooms should not be shared between reactors. Each cable spreading room should be separated from other areas of the plant by barriers (walls and floors) having a minimum fire resistance of three hours. Cabling for redundant safety divisions should be separated by walls having three hour fire barriers.

Alternative guidance for constructed plants is shown in Section F.3, “Cable Spreading Room”.

CPNPP Fire Protection Program

Two cable spreading rooms are included in the design of CPNPP, one for each unit. These rooms are separated by three hour rated fire barriers except as noted in Reference [19]. See Section F.3 for the design description.

APCSB 9.5-1 Appendix A

D.1.d Interior wall and structural components, thermal insulation materials and radiation shielding materials and sound-proofing should be non-combustible. Interior finishes should be non-combustible or listed by a nationally recognized testing laboratory, such as Factory Mutual or Underwriters’ Laboratory, Inc. for flame spread, smoke and fuel contribution of 25 or less in its use configuration (ASTM E-84 Test “Surface Burning Characteristics of Building Materials”).

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline except as noted in section 9.5.1.6.2. Specific criteria of the structural and interior construction materials are described in Subsection 9.5.1.5.1, 9.5.1.5.2, and 9.5.1.5.3.

APCSB 9.5-1 Appendix A

D.1.e Metal deck roof construction should be non-combustible (see the building materials directory of the Underwriters' Laboratory, Inc.) or listed as Class I by Factory Mutual System Approval Guide.

CPNPP Fire Protection Program

Metal roof deck construction is not used at CPNPP for power block or other safety-related buildings.

APCSB 9.5-1 Appendix A

D.1.f Suspended ceilings and their supports should be of non-combustible construction. Concealed spaces should be devoid of combustibles.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. The use of combustible material is minimized. Also see Guideline D.3.j. See Subsection 9.5.1.5.3.1 for specific criteria.

APCSB 9.5-1 Appendix A

D.1.g High voltage - high amperage transformers installed inside buildings containing safety related systems should be of the dry type or insulated and cooled with non-combustible liquid.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. See Subsection 9.5.1.5.6 for specific criteria.

APCSB 9.5-1 Appendix A

D.1.h Buildings containing safety related systems, having openings in exterior walls closer than 50 feet to flammable oil filled transformers should be protected from the effects of a fire by:

(i) closing of the opening to have fire resistance equal to three hours,

(ii) constructing a three hour fire barrier between the transformers and the wall openings; or

(iii) closing the opening and providing the capability to maintain a water curtain in case of a fire.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline except as noted in reference [19]. See Subsection 9.5.1.5.6 for specific criteria.

APCSB 9.5-1 Appendix A

D.1.i Floor drains, sized to remove expected fire fighting water flow should be provided in those areas where fixed water fire suppression systems are installed. Drains should also be provided in other areas where hand hose lines may be used if such fire fighting water could cause unacceptable damage to equipment in the area. Equipment should be installed on pedestals, or curbs should be provided as required to contain water and direct it to floor drains. (See NFPA 92M, "Waterproofing and Draining of Floors.") Drains in areas containing combustible liquids should have provisions for preventing the spread of the fire throughout the drain system. Water drainage from areas which may contain radioactivity should be sampled and analyzed before discharge to the environment.

In operating plants or plants under construction, if accumulation of water from the operation of new fire suppression systems does not create unacceptable consequences, drains need not be installed.

CPNPP Fire Protection Program

Floor drains are provided in areas protected by fixed water suppression systems. Drainage of water from areas that are not serviced by the floor drain systems is accomplished by using both the floor drains in the corridors adjacent to the respective fire areas and by manual means after a fire. The switchgear and electrical penetration areas, and the Control Room are not provided with floor drains. If a water sprinkler system discharges or if firefighting with hoses is necessary, doors leading into these areas will be opened and the water will run out into adjacent corridor floor drains. In those areas where electrical cabinets are installed, drainage paths are sufficient to prevent excessive water levels. The battery rooms or their adjacent corridors are provided with floor drains. The cable spreading rooms are provided with floor drains. Drainage of areas where combustible liquids are stored in tanks are designed to prevent the spread of a fire through the drain system. The only safety related areas to which this applies is the DG day tank rooms

which do not have floor drains. See Section 9.5.4 for additional details. Drainage from areas that may contain radioactive material is monitored prior to discharge outside the plant.

APCSB 9.5-1 Appendix A

D.1.j Floors, walls and ceilings enclosing separate fire areas should have a minimum fire rating of three hours. Penetrations in these fire barriers, including conduits and piping, should be sealed or closed to provide a fire resistance rating at least equal to that of the fire barrier itself. Door openings should be protected with equivalent rated doors, frames and hardware that have been tested and approved by a nationally recognized laboratory. Such doors should be normally closed and locked or alarmed with alarm and annunciation in the control room. Penetrations for ventilation system should be protected by a standard “fire door damper” where required. (Refer to NFPA 80, “Fire Doors with Windows”). The fire hazard in each area should be evaluated to determine barrier requirements. If barrier fire resistance cannot be made adequate, fire detection and suppression should be provided, such as:

(i) water curtain in case of fire,

(ii) flame retardant coatings,

(iii) additional fire barriers.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline except as noted in Section 9.5.1.6.2 and reference [19]. The barriers, doors, and penetrations are discussed in the applicable sections of Subsection 9.5.1.5. Fire doors located in fire barriers are normally closed or are automatically closed by listed closure devices and released by solenoids, thermal-links, or electro-thermal links. Where doors have been modified for security purposes, the modifications have been accepted by United Laboratory Inc.

Locks and alarms are provided on doors in accordance with the plant fire protection and security program. The fire barrier resistance rating is generally based on the amount of combustible material in the area.

APCSB 9.5-1 Appendix A

D.2 Control of Combustibles

D.2.a Safety-related systems should be isolated or separated from combustible materials. When this is not possible because of the nature of the safety system or the combustible material, special protection should be provided to prevent a fire from defeating the safety system function. Such protection may involve a combination of automatic fire suppression, and construction capable of withstanding and containing a fire that consumes all combustibles present. Examples of such combustible materials that may not be separable from the remainder of its system are:

(1) Emergency diesel generator fuel oil day tanks

(2) Turbine-generator oil and hydraulic control fluid systems

(3) Reactor coolant pump lube oil system.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. Separation and protection of required fire safe shutdown systems are discussed for each fire area in the Fire Protection Report.

APCSB 9.5-1 Appendix A

D.2.b Bulk gas storage (either compressed or cryogenic), should not be permitted inside structures housing safety related equipment. Storage of flammable gas such as hydrogen, should be located outdoors or in separate detached buildings so that a fire or explosion will not adversely affect any safety related systems or equipment.

Care should be taken to locate high pressure gas storage containers with the long axis parallel to building walls. This will minimize the possibility of wall penetration in the event of a container failure. Use of compressed gases (especially flammable and fuel gases) inside buildings should be controlled. (Refer to NFPA 6, “Industrial Fire Loss Prevention.”)

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline except as noted in Section 9.5.1.6.2. Bulk gas storage is discussed in Section 9.5.1.5.7.

APCSB 9.5-1 Appendix A

D.2.c The use of plastic materials should be minimized. In particular, halogenated plastics such as polyvinyl chloride (PVC) and neoprene should be used only when substitute non-combustible materials are not available. All plastic materials, including flame and fire retardant materials, will burn with an intensity and BTU production in a range similar to that of ordinary hydrocarbons. When burning, they produce heavy smoke that obscures visibility and can plug air filters, especially charcoal and HEPA. The halogenated plastics also release free chlorine and hydrogen chloride when burning which are toxic to humans and corrosive to equipment.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. Plastic material is used when required for radiation resistance reasons. Halongenated plastics such as polyvinyl chloride

(PVC) are not used as an exposed cable insulation or jacketing material at CPNPP except as discussed in Subsection D.3.f.

APCSB 9.5-1 Appendix A

D.2.d Storage of flammable liquids should, as a minimum, comply with the requirements of NFPA 30, “Flammable and Combustible Liquids Code.”

CPNPP Fire Protection Program

The recommendations of NFPA 30 for storage of flammable and combustible liquids were used as guidance in the development of CPNPP plant procedures.

APCSB 9.5-1 Appendix A

D.3 Electric Cable Construction, Cable Trays and Cable Penetrations

D.3.a Only non-combustible materials should be used for cable tray construction.

CPNPP Fire Protection Program

The Fire Protection Program complies with the guideline. All cable tray construction materials are noncombustible.

APCSB 9.5-1 Appendix A

D.3.b See Section E.3 for fire protection guidelines for cable trays outside the cable spreading room.

CPNPP Fire Protection Program

Criteria are discussed and referenced in Subsection 9.5.1.5.5 and guideline D.3.f.

APCSB 9.5-1 Appendix A

D.3.c Automatic water sprinkler systems should be provided for cable trays outside the cable spreading room. Cables should be designed to allow wetting down with deluge water without electrical faulting. Manual hose stations and portable hand extinguishers should be provided as backup. Safety related equipment in the vicinity of such cable trays, that does not itself require water fire protection, but is subject to unacceptable damage from sprinkler water discharge, should be protected from sprinkler system operation or malfunction.

CPNPP Fire Protection Program

Automatic and manual pre-action water sprinklers are provided for cable trays outside containment and cable spreading rooms where a congestion of cable trays exist (i.e., 4 or more stacked cable trays). Manual hose stations are the primary suppression for cable trays where the following conditions are met:

1. Low cable tray congestion (cable trays are not stacked greater than three high).

2. Cable trays that contain safe shutdown cabling are separated from their redundant trains by three hour rated fire barriers.

3. Smoke detectors are provided in the area of the cables.

Cables are designed to allow wetting down without electrical faulting. Manual hose stations and portable extinguishers are provided as backup. Essential equipment that could be damaged by a sprinkler system discharge is protected. Also see Guideline A.5.

APCSB 9.5-1 Appendix A

D.3.d Cable and cable tray penetration of fire barriers (vertical and horizontal) should be sealed to give protection at least equivalent to that fire barrier. The design of fire barriers for horizontal and vertical cable trays should, as a minimum, meet the requirements of ASTM E-119, “Fire Test of Building Construction and Materials,” including the hose stream test.

Alternate criteria: Where installed penetration seals are deficient with respect to fire resistance, these seals may be protected by covering both sides with an approved fire retardant material. The adequacy of using such material should be demonstrated by suitable testing.

CPNPP Fire Protection Program

For conduits which are greater than four (4) inches nominal size, internal seals are installed either at the barrier or on both sides of the barrier at the first opening in the direction of the barrier. These internal seals have a fire rating equal to or greater than that of the fire barrier rating.

For conduits which are less than or equal to four (4) inches nominal size, and automatic suppression and (manually actuated suppression for switchgear and adjacent rooms in Unit 2) detection are provided on both sides of the barrier, internal seals are installed in the barrier; gas and smoke seals are installed at the first opening on both sides of the barrier; or internal seals are installed at the first opening on either side of the barrier with a fire rating equivalent to that of the barrier, unless individually evaluated and documented in Section 9.5.1.6.2 or Reference [19]. For conduits which are less than or equal to four (4) inches nominal size, and automatic suppression (manually actuated suppression for switchgear and adjacent rooms in Unit 2) and detection are not provided on both sides of the barrier, internal seals are installed at the barrier

with a fire rating equivalent to that of the barrier, or gas and smoke seals are installed at the first opening on both sides of the barrier, except as described in 9.5.1.6.2 or Reference [19].

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D.3.e Fire breaks should be provided as deemed necessary by the fire hazards analysis. Flame or flame retardant coatings may be used as a fire break for grouped electrical cables to limit spread of fire in cable ventings. (Possible cable derating owing to use of such coating materials must be considered during design.)

CPNPP Fire Protection Program

The Fire Protection Program complies with the guideline. Fire breaks are provided as described in Subsection 9.5.1.5.5.

APCSB 9.5-1 Appendix A

D.3.f Electric cable constructions should as a minimum pass the current IEEE No. 383 flame test. (This does not imply that cables passing this test will not require additional fire protection).

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with this guideline except as noted in section 9.5.1.6.2. Electrical cable construction is described in Section 9.5.1.5.5.

APCSB 9.5-1 Appendix A

D.3.g To the extent practical, cable construction that does not give off corrosive gases while burning should be used.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guidelines to the extent practical with present day cable insulation and jacket material.

APCSB 9.5-1 Appendix A

D.3.h Cable trays, raceways, conduit, trenches, or culverts should be used only for cables. Miscellaneous storage should not be permitted, nor should piping for flammable or combustible liquids or gases be installed in these areas.

CPNPP Fire Protection Program

The Fire Protection Program complies with the guideline.

APCSB 9.5-1 Appendix A

D.3.i The design cable tunnels, culverts and spreading rooms should provide for automatic or manual smoke venting as required to facilitate manual fire fighting capability.

CPNPP Fire Protection Program

The Fire Protection Program provides for manual smoke venting to enable manual fire fighting. See Subsection 9.5.1.5.4 for criteria on smoke venting.

APCSB 9.5-1 Appendix A

D.3.j Cables in the control room should be kept to the minimum necessary for operation of the control room. All cables entering the control room should terminate there. Cables should not be installed in floor trenches or culverts in the control room.

CPNPP Fire Protection Program

The cables in the Control Room are the minimum necessary for operation of the plant. There are no cables routed in floor trenches in the Control Room. There is, however, a small amount of cabling enclosed in steel conduit, routed above the suspended ceiling in the Control Room. Fire detection is provided for this concealed area.

APCSB 9.5-1 Appendix A

D.4 Ventilation

D.4.a The products of combustion that need to be removed from a specific fire area should be evaluated to determine how they will be controlled. Smoke and corrosive gases should generally be automatically discharged directly outside to a safe location. Smoke and gases containing radioactive materials should be monitored in the fire area to determine if release to the environment is within the permissible limits of the Offsite Dose Calculation Manual.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. See Section 9.5.1.5.4 for further information.

APCSB 9.5-1 Appendix A

D.4.b Any ventilation system designed to exhaust smoke or corrosive gases should be evaluated to ensure that inadvertent operation or single failures will not violate the controlled areas of the plant design. This requirement includes containment functions for protection of the public and maintaining habitability for operations personnel.

CPNPP Fire Protection Program

Ventilation of smoke or corrosive gases, resulting from a fire will be accomplished manually as required, and subsequent to monitoring for radioactivity and evaluating the products of combustion.

APCSB 9.5-1 Appendix A

D.4.c The power supply and controls for mechanical ventilation systems should be run outside the fire area served by the system.

CPNPP Fire Protection Program

The Fire Protection Program complies with the guideline. The power supplies for smoke ejectors are described in Subsection 9.5.1.5.4.

APCSB 9.5-1 Appendix A

D.4.d Fire suppression systems should be installed to protect charcoal filters in accordance with Regulatory Guide 1.52, “Design Testing and Maintenance Criteria for Atmospheric Cleanup Air Filtration.”

CPNPP Fire Protection Program

The Fire Protection Program complies with the guideline. All HVAC charcoal absorber beds are provided with integral water spray systems. These systems are described in Subsection 9.5.1.4.2.c.

APCSB 9.5-1 Appendix A

D.4.e The fresh air supply intakes to areas containing safety related equipment or systems should be located remote from the exhaust air outlets and smoke vents of other fire areas to minimize the possibility of contaminating the intake air with the products of combustion.

CPNPP Fire Protection Program

The Fire Protection Program complies with the guideline. The use of combustible materials is minimized. In general, ventilation air intakes are located on the exterior walls and the outlets are located on the roof.

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D.4.f Stairwells should be designed to minimize smoke infiltration during a fire. Staircases should serve as escape routes and access routes for fire fighting. Fire exit routes should be clearly marked. Stairwells, elevators and chutes should be enclosed in masonry towers with minimum fire rating of three hours and automatic fire doors at least equal to the enclosure construction, at each opening into the building. Elevators should not be used during fire emergencies.

CPNPP Fire Protection Program

The Fire Protection Program complies with this guidance except as noted in section 9.5.1.6.2. Walls enclosing stairwells are constructed of concrete and gypsum dry walls and are provided with Class B labeled automatic fire doors. All enclosed stairwells are pressurized to minimize smoke infiltration.

APCSB 9.5-1 Appendix A

D.4.g Smoke and heat vents may be useful in specific areas such as cable spreading rooms and diesel fuel oil storage areas and switchgear rooms. When natural-convection ventilation is used, a minimum ratio of 1 sq. foot of venting area per 200 sq. feet of floor area should be provided. If forced-convection ventilation is used, 300 CFM should be provided for every 200 sq. feet of floor area. See NFPA No. 204 for additional guidance of smoke control.

CPNPP Fire Protection Program

Smoke and heat venting will be accomplished manually as required. See Section 9.5.1.5.4 for discussion.

APCSB 9.5-1 Appendix A

D.4.h Self-contained breathing apparatus, using full face positive pressure masks, approved by NIOSH (National Institute for Occupational Safety and Health - approval formerly given by the U.S. Bureau of Mines) should be provided for fire brigade, damage control and control room personnel. Control room personnel may be furnished breathing air by a manifold system piped from a storage reservoir if practical. Service of operating life should be a minimum of one-half hour for the self-contained units.

CPNPP Fire Protection Program

Self-contained breathing apparatus are provided for fire brigade, damage control, and control room personnel. Service life for the self contained units exceed one-half hour.

APCSB 9.5-1 Appendix A

D.4.h At least two extra air bottles should be located onsite for each self-contained breathing unit. In addition, an onsite 6-hour supply of reserve air should be provided and arranged to permit quick and complete replenishment of exhausted supply air bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air should be used. Special care must be taken to locate the compressor in areas free of dust and contaminants.

CPNPP Fire Protection Program

The Fire Protection Program complies with this guideline.

APCSB 9.5-1 Appendix A

D.4.i Where total flooding gas extinguishing systems are used, area intake and exhaust ventilation dampers should close upon initiation of gas flow to maintain necessary gas concentration. (See NFPA 12, “Carbon Dioxide Systems”, and 12A, “Halon 1301 Systems.”)

CPNPP Fire Protection Program

The Fire Protection Program complies with the guideline. See Subsection 9.5.1.5.4 for description of fire dampers.

APCSB 9.5-1 Appendix A

D.5 Lighting and Communication

Lighting and two way voice communication are vital to safe shutdown and emergency response in the event of fire. Suitable fixed and portable emergency lighting and communication devices should be provided to satisfy the following requirements:

(a) Fixed emergency lighting should consist of sealed beam units with individual 8-hour minimum battery power supplies.

(b) Suitable sealed beam battery powered portable hand lights should be provided for emergency use.

CPNPP Fire Protection Program

Areas containing fire safe shutdown equipment required to achieve hot standby, and primary interior egress and access routes between these areas, are provided with DC Emergency Lighting supplied by 8 hour sealed beam or fluorescent lamp battery power pack units (except in

the Control Room). DC Emergency Lighting in the Control Room is supplied power from the non-class 1E dedicated 8-hour batteries. (See FSAR Section 9.5.3.2.1 and 9.5.1.6.2). Supplemental lighting is provided from battery-powered hand held portable lights.

APCSB 9.5-1 Appendix A

D.5 (c) Fixed emergency communication should use voice powered head sets at pre-selected stations.

CPNPP Fire Protection Program

The Fire Protection Program provides intra plant portable radio with page-party/public address system backup for use in emergency conditions instead of voice powered head sets. For additional description of the communication systems, see Subsection 9.5.2.2.

APCSB 9.5-1 Appendix A

D.5 (d) Fixed repeater installed to permit use of portable radio communication units should be protected from exposure to fire damage.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with this guideline by providing radio to radio "talkaround" and plant page party/public address system capability in the event of fire damage to the repeater.

APCSB 9.5-1 Appendix A

E. Fire Detection and Suppression

E.1 Fire Detection

E.1.a Fire detection systems should as a minimum comply with NFPA 72D, "Standard for the Installation, Maintenance and Use of Proprietary Protective Signaling Systems."

CPNPP Fire Protection Program

The Fire Protection Program complies with this guideline set forth by the NRC Fire detection system design parameters are described in Subsection 9.5.1.4.2. Location and placement of detectors are in accordance with the guidelines of NFPA 72E except where special conditions did not permit. In such cases a Fire Protection Engineer located the detector based on engineering judgement as permitted by NFPA 72E or provided justification per reference [19].

APCSB 9.5-1 Appendix A

E.1.b Fire detection system should give audible and visual alarm and annuncciation in the control room. Local audible alarms should also sound at the location of the fire.

CPNPP Fire Protection Program

Fire detection systems give audible and visual alarms and annunciation in the Control Room. Local audible alarms sound at the local detector control panel. Fire protection procedures provide for the use of the Public Address System (Gaitronics) for audible annunciation of the fire alarm location throughout the plant upon receipt of an alarm in the Control Room except in the case of a fire in the Control Room complex. A separate alarm is provided for audible annunciation of a fire in the Control Room complex. The Gaitronics will be maintained operational at all times with a minimum reasonable outage duration for maintenance or repair.

APCSB 9.5-1 Appendix A

E.1.c Fire alarms should be distinctive and unique. They should not be capable of being confused with any other plant system alarms.

CPNPP Fire Protection Program

Fire alarms are distinctive and unique and will not be confused with other plant alarms.

APCSB 9.5-1 Appendix A

E.1.d Fire detection and actuation systems should be connected to the plant emergency power supply.

CPNPP Fire Protection Program

Fire detection and actuation systems are connected to plant emergency power supplies in accordance with NFPA 72D Section 2220.

APCSB 9.5-1 Appendix A

E.2 Fire Protection Water Supply Systems

E.2.a An underground yard fire main loop should be installed to furnish anticipated fire water requirements. NFPA 24 - “Standard for Outside Protection” - gives necessary guidance for such installation. It references other design codes and standards developed by such organizations as the American National Standards Institute (ANSI) and the American Water Works Association (AWWA). Lined steel or cast iron pipe should be used to reduce internal tuberculation. Such tuberculation deposits in an unlined pipe over a period of years can significantly reduce water flow through the combination of increased friction and reduced pipe diameter. Means for treating and flushing the systems should be provided. Approved visually indicating sectional control valves, such as Post Indicator Valves, should be provided to isolate portions of the main for maintenance or repair without shutting off the entire system.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. The system is described in Subsection 9.5.1.4.

APCSB 9.5-1 Appendix A

E.2.a The fire main system piping should be separate from service or sanitary water system piping.

CPNPP Fire Protection Program

CPNPP is in compliance with the guideline

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E.2.b A common yard fire main loop may serve multi-unit nuclear power plant sites, if cross-connected between units. Section control valves should permit maintaining independence of the individual loop around each unit. For such installations, common water supplies may also be utilized. The water supply should be sized for the largest single expected flow. For multiple reactor sites with widely separated plants (approaching 1 mile or more), separate yard fire main loops should be used.

CPNPP Fire Protection Program

A common yard fire main loop is provided for the entire CPNPP site. Sectional control valves are provided for isolation of sections and to ensure at least one flow path to the individual branch lines provided for various buildings with any one section isolated. This arrangement eliminates the requirement for cross connection between units. See Subsection 9.5.1.4 for a description of the sizing of the common water supply.

APCSB 9.5.1 Appendix A

E.2.c If pumps are required to meet system pressure or flow requirements, a sufficient number of pumps should be provided so that 100 percent capacity will be available with one pump inactive (e.g., three 50 percent pumps or two 100 percent pumps). The connection to the yard fire main loop from each fire pump should be widely separated, preferably located on opposite sides of the plant. Each pump should have its own driver with independent power supplies and control. At least one pump (if not powered from the emergency diesels) should be driven by non-electrical means, preferably diesel engine. Pumps and drivers should be located in rooms separated from the remaining pumps and equipment by a minimum three hour fire wall. Alarms indicating pump running, driver availability, or failure to start should be provided in the control room.

Details of the fire pump installation should as a minimum conform to NFPA 20, “Standard for the Installation of Centrifugal Fire Pumps.”

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. Three 50 percent capacity pumps are provided. Two of the three pumps are diesel engine-driven pumps and are located in separate fire compartments in the pump house. The other fire pump is an electric motor-driven pump and is located in a separate fire compartment in the pump house. An independent connection to the loop is provided for each pump. See Subsection 9.5.1.4 for further details concerning the pumps.

APCSB 9.5-1 Appendix A

E.2.d Two separate reliable water supplies should be provided. If tanks are used, two 100 percent (minimum of 300,000 gallons each) system capacity tanks should be installed. They should be so interconnected that pumps can take suction from either or both. However, a leak in one tank or its piping should not cause both tanks to drain. The main plant fire water supply capacity should be capable of refilling either tank in a minimum of eight hours.

Common tanks are permitted for fire and sanitary or service water storage. When this is done, however, minimum fire water storage requirements should be dedicated by means of a vertical standpipe for other water services.

CPNPP Fire Protection Program

Two 100 percent capacity atmospheric fire water storage tanks are provided, each with a nominal capacity of 524,500 gallons. The tanks are interconnected to facilitate suction from either or both tanks. Refill capability with a separate pump, which takes suction from the safe shutdown impoundment (SSI), is provided to allow either tank to be refilled within 8 hours after using its contents to extinguish a fire.

APCSB 9.5-1 Appendix A

E.2.e The fire water supply (total capacity and flow rate) should be calculated on the basis of the largest expected flow rate for a period of two hours, but not less than 300,000 gallons. This flow rate should be based (conservatively) on 1,000 gpm for manual hose streams plus the greater of:

(1) all sprinkler heads opened and flowing in the largest designed fire area; or

(2) the largest open head deluge system(s) operating.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline except as noted in section 9.5.1.6.2. The fire water supply capacity and flow rate is described in Subsection 9.5.1.4.2

APCSB 9.5-1 Appendix A

E.2.f Lakes or fresh water ponds of sufficient size may qualify as the sole source of water for fire protections, but requires at least two intakes to the pump supply. When a common water supply is permitted for fire protection and the ultimate heat sink, the following conditions should also be satisfied:

(1) The additional fire protection water requirements are designed into the total storage capacity; and

(2) Failure of the fire protection system should not degrade the function of ultimate heat sink.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. CPNPP employs two storage tanks dedicated to fire protection use for water supply requirements.

APCSB 9.5-1 Appendix A

E.2.g Outside manual hose installation should be sufficient to reach any location with an effective hose stream. To accomplish this, hydrants should be installed approximately every 250 feet on the yard main system. The lateral to each hydrant from the yard main should be controlled by a visually indicating or key operated (curb) valve. A hose house, equipped with hose and combination nozzle, and other auxiliary equipment recommended in NFPA 24, “Outside Protection,” should be provided as needed but at least every 1,000 feet.

Threads compatible with those used by local fire departments should be provided on all hydrants, hose couplings and standpipe risers.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. Auxiliary gate valves (curb box valves) are provided adjacent to each hydrant in accordance with ANI criteria. A description of the outside hose station layout, auxiliary equipment contained in each cabinet and the type of threads on all fire fighting equipment is provided in Subsection 9.5.1.4.

APCSB 9.5-1 Appendix A

E.3 Water Sprinklers and Hose Standpipe Systems

E.3.a Each Automatic sprinkler system and manual hose station standpipe should have an independent connection to the plant underground water main. Headers fed from each end are permitted inside buildings to supply multiple sprinkler and standpipe systems. When provided, such headers are considered an extension of the yard main system. The header arrangement should be such that no single failure can impair both the primary and backup fire protection systems.

Each sprinkler and standpipe system should be equipped with OS&Y (outside screw and yoke) gate valve, or other approved shut off valve, and water flow alarm. Safety related equipment that does not itself require sprinkler water fire protection, but is subject to unacceptable damage if wetted by sprinkler water discharge should be protected by water shields or baffles.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline except as noted in Section 9.5.1.6.2. The connection of each automatic sprinkler system and hose station is described in Subsection 9.5.1.4.

Also see guideline D.3.c.

APCSB 9.5-1 Appendix A

E.3.b All valves in the fire water systems should be electrically supervised. The electrical supervision signal should indicate in the control room and other appropriate command locations in the plant (See NFPA 26, “Supervision of Valves”).

Alternate Criteria: When electrical supervision of fire protection valves is not practicable, an adequate management supervision program should be provided. Such a program should include locking valves open with strict key control; tamper proof seals; and periodic, visual check of all valves.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the criteria. All isolation or sectional control valves are electrically supervised, or locked in the appropriate position. Appropriate procedures will be developed for key control and for periodic visual inspection of valves.

APCSB 9.5-1 Appendix A

E.3.c Automatic sprinkler systems should as a minimum conform to requirements of appropriate standards such as NFPA 13, “Standard for the Installation of Sprinkler Systems”, and NFPA 15, “Standard for Water Spray Fixed Systems.”

CPNPP Fire Protection Program

Automatic sprinkler systems comply with requirements of NFPA 13 and NFPA 15. Water spray systems for the charcoal filters are designed to the requirements of Regulatory Guide 1.52.

Design densities for cable tray suppression systems located in congested cable areas meet the applicable NFPA Standards (Reference 24). Specific differences with the applicable NFPA standards are identified and justified in Reference [19].

APCSB 9.5-1 Appendix A

E.3.d Interior manual hose installation should be able to reach any location with at least one effective hose stream. To accomplish this, standpipes with hose connections, equipped with a maximum of 75 feet of 1-1/2 inch woven jacket lined fire hose and suitable nozzles should be provided in all buildings, including containment, on all floors and should be spaced at not more than 100 foot intervals. Individual standpipes should be of at least 4-inch diameter for multiple hose connections and 2-1/2-inch diameter for single hose connections. These systems should follow the requirements of NFPA 14 for sizing, spacing and pipe support requirements (NELPIA).

Hose stations should be located outside entrances to normally unoccupied areas and inside normally occupied areas. Standpipes serving hose stations in areas housing safety related equipment should have shut off valves and pressure reducing devices (if applicable) outside the area.

CPNPP Fire Protection Program

NFPA 14 was used as guidance for installation of Class II service interior manual hose stations. Each hose station is equipped with 100 feet of 1-1/2 inch woven jacket lined fire hose and a nozzle compatible with the type of fire postulated. The spacing of the hose stations ensures that at least one effective hose stream can reach any location in safety-related areas of the plant except where identified and justified in the Fire Protection Report, Reference [19]. NFPA 14 was used as guidance for sizing of Class II type standpipes and hose systems. For a further description of the interior hose stations see Subsection 9.5.1.4.

APCSB 9.5-1 Appendix A

E.3.e The proper type of hose nozzles to be supplied to each area should be based on the fire hazard analysis. The usual combination spray/straight-stream nozzle may cause unacceptable mechanical damage (for example, the delicate electronic equipment in the control room) and be unsuitable. Electrically safe nozzles should be provided at locations where electrical equipment or cabling is located.

CPNPP Fire Protection Program

FOG type nozzles are provided in hose cabinets inside plant buildings. Outside hose houses are provided with combination nozzles.

APCSB 9.5-1 Appendix A

E.3.f Certain fires such as those involving flammable liquids respond well to foam suppression. Consideration should be given to use of any of the available foams for such specialized protection application. These include the more common chemical and mechanical low expansion foams, high expansion foam and the relatively new Aqueous Film Forming Foam (AFFF).

CPNPP Fire Protection Program

No fixed foam systems are used at the plant but manual foam-based fire fighting equipment and training is provided.

APCSB 9.5-1 Appendix A

E.4 Halon Suppression Systems

The use of Halon fire extinguishing agents should as a minimum comply with the requirements of NFPA 12A and 12B, “Halogenated Fire Extinguishing Agent Systems - Halon 1301 and Halon 1211.” Only UL or FM approved agents should be used.

In addition to the guidelines of NFPA 12A and 12B, preventative maintenance and testing of the systems, including check weighing of the Halon cylinders should be done at least quarterly.

Particular consideration should also be given to:

(a) minimum required Halon concentration and soak time;

(b) toxicity of Halon;

(c) toxicity and corrosive characteristics of thermal decomposition products of Halon.

CPNPP Fire Protection Program

The CPNPP Fire Protection Program uses NFPA 12A and 12B, “Halogenated Fire Extinguishing Agent Systems - Halon 1301 and 1211,” as a guideline for the halon suppression system design and installation. Design parameters and a system description are stated in Subsection 9.5.1.4.2. Administrative controls for Halon systems are in accordance with Reference [19].

APCSB 9.5-1 Appendix A

E.5 Carbon Dioxide Suppression Systems

The use of carbon dioxide extinguishing systems should as a minimum comply with the requirements of NFPA 12, "Carbon Dioxide Extinguishing Systems."

Particular consideration should also be given to:

(a) minimum required CO_2 concentration and soak time;

(b) toxicity of CO_2;

(c) possibility of secondary thermal shock (cooling) damage;

(d) offsetting requirements for venting during CO_2 injection to prevent overpressurization versus sealing to prevent loss of agent;

(e) design requirements from overpressurization; and

(f) possibility and probability of CO_2 systems being out of service because of personnel safety consideration. CO_2 systems are disarmed whenever people are present in an area so protected. Areas entered frequently (even though duration time for any visit is short) have often been found with CO_2 systems shut off.

CPNPP Fire Protection Program

The Fire Protection Program does not use fixed carbon dioxide suppression systems.

APCSB 9.5-1 Appendix A

E.6 Portable Extinguishers

Fire extinguishers should be provided in accordance with guidelines of NFPA 10 and 10A, "Portable Fire Extinguishers Installation, Maintenance and Use." Dry chemical extinguishers should be installed with due consideration given to cleanup problems after use and possible adverse effects on equipment installed in the area.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. See Subsection 9.5.1.4.2 for description of portable fire extinguishers.

APCSB 9.5-1 Appendix A

F. Guidelines for Specifics Plant Areas

F.1 Primary and Secondary Containment

F.1.a Normal Operation

Fire protection requirements for the primary and secondary containment areas should be provided on the basis of specific identified hazards. For example:

- Lubricating oil or hydraulic fluid system for the primary coolant pumps
- Cable tray arrangements and cable penetrations
- Charcoal filters

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with this guideline except as noted in Section 9.5.1.6.2. The Fire Protection Report identifies specific hazards inside containment. Fire detection and suppression are provided accordingly.

APCSB 9.5-1 Appendix A

F.1.a Fire suppression systems should be provided based on the fire hazards analysis.

Fixed fire suppression capability should be provided for hazards that could jeopardize safe plant shutdown. Automatic sprinklers are preferred. An acceptable alternate is automatic gas (Halon or CO_2) for hazards identified as requiring fixed suppression protection.

An enclosure may be required to confine the agent if a gas system is used. Such enclosures should not adversely affect safe shutdown, or other operating equipment in containment.

CPNPP Fire Protection Program

Automatic water spray systems are provided for the carbon absorber beds of the pre-access filter units located inside the Containment. See Section 9.5.1.4.2 for description of these systems. The reactor coolant pumps are equipped with an oil collection system. This system is designed to collect and drain, to a safe place, any oil which may be discharged from the reactor coolant pump lubrication system.

APCSB 9.5-1 Appendix A

F.1.a Operation of the fire protection systems should not compromise integrity of the containment or the other safety related systems. Fire protection activities in the containment areas should function in conjunction with total containment requirements such as control of contaminated liquid and gaseous release and ventilation.

CPNPP Fire Protection Program

Operation of the fire protection system does not compromise the integrity of the Containment.

APCSB 9.5-1 Appendix A

F.1.a Fire detection systems should alarm and annunciate in the control room. The type of detection used and the location of the detectors should be most suitable to the particular type of fire that could be expected from the identified hazard. A primary containment general area fire detection capability should be provided as backup for the above described hazard detection. To accomplish this, suitable smoke detection (e.g., visual obscuration, light scattering and particle counting) should be installed in the air recirculation system ahead of any filters.

Automatic fire suppression capability need not be provided in the primary containment atmospheres that are inserted during normal operation. However, special fire protection requirements during refueling and maintenance operations should be satisfied as provided below.

CPNPP Fire Protection Program

Fire detection is provided throughout the Containment in accordance with the results of Fire Hazard Analysis Evaluations, which are summarized in the Fire Protection Report. Alarms are provided in the Control Room to annunciate a fire condition via the plant public address (PA) system.

APCSB 9.5-1 Appendix A

F.1.b Refueling and Maintenance

Refueling and maintenance operations in containment may introduce additional hazards such as contamination control materials, decontamination supplies, wood planking, temporary wiring, welding and flame cutting (with portable compressed fuel gas supply). Possible fires would not necessarily be in the vicinity of fixed detection and suppression systems.

Management procedures and controls necessary to assure adequate fire protection are discussed in Section 3a.

In addition, manual fire fighting capability should be permanently installed in containment. Standpipes with hose stations, and portable fire extinguishers, should be installed at strategic locations throughout containment for any required manual fire fighting operations.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. Standpipes and hose stations are located on each elevation in the Containment Buildings such that an effective hose stream can reach any location, except as identified and justified in Reference [19]. Portable extinguishers are provided for each Containment Building. To reduce radiation related deterioration, portable extinguishers and hoses may be removed from the containment during

normal operation. Whenever the containment is to be occupied for longer than 48 hours, the portable extinguishers and hoses are returned to their designated locations in containment.

APCSB 9.5-1 Appendix A

F.1.b. Adequate self-contained breathing apparatus should be provided near the containment entrances for fire fighting and damage control personnel. These units should be independent of any breathing apparatus or air supply systems provided for general plant activities.

CPNPP Fire Protection Program

Adequate, self-contained breathing apparatus are provided at the fire brigade staging area located in the turbine building. By procedure the fire brigade gathers all required equipment from this location for any fire in the plant including containment fires. This location is sufficiently near the containment to provide for prompt action. Additional air cylinders are located at the containment entrance.

APCSB 9.5-1 Appendix A

F.2 Control Room

(a) The Control Room is essential to safe reactor operation. It must be protected against disabling fire damage and should be separated from other areas of the plant by floors, walls and roofs having minimum fire resistance ratings of three hours.

(b) Exposure fire involving combustibles in the general room area.

CPNPP Fire Protection Program

The Fire Protection Program is in general compliance with the guideline, except as noted in FSAR Section 9.5.1.6.2. The Control Room is separated from other areas of the plant by three hour rated fire barriers, except as noted in Reference [19].

APCSB 9.5-1 Appendix A

F.2 Control Room cabinets and consoles are subject to damage from two distinct fire hazards:

(a) Fire originating within a cabinet or console; and

(b) Exposure fire involving combustibles in the general room area.

Manual fire fighting capability should be provided for both hazards.

CPNPP Fire Protection Program

Manual fire fighting capability employing portable water and Halon extinguishers and hose stations are provided inside the Control Room.

APCSB 9.5-1 Appendix A

F.2 Hose stations and portable water and Halon extinguishers should be located in the control room to eliminate the need for operators to leave the control room. An additional hose piping shut off valve and pressure reducing device should be installed outside the control room.

Hose stations adjacent to the control room with portable extinguishers in the control room are acceptable.

Nozzles that are compatible with the hazards and equipment in the control room should be provided for the manual hose station. The nozzles chosen should satisfy actual fire fighting needs, satisfy electrical safety and minimize physical damage to electrical equipment from hose stream impingement.

Fire detection in the control room cabinets, and consoles should be provided by smoke and heat detectors in each fire area. Alarm and annunciation should be provided in the control room. Fire alarms in other parts of the plant should also be alarmed and annunciated in the control room.

CPNPP Fire Protection Program

Manual fire fighting capability employing portable water and Halon extinguishers, as well as hose stations, is provided in the Control Room.

Fire detection is provided in each console in the Control Room except as noted in Section 9.5.1.6.2.

The Fire Detection Main Control Panel is located in the Control Room. Any fire alarm actuated anywhere in the plant is indicated audibly and visually on this panel.

APCSB 9.5-1 Appendix A

F.2 Breathing apparatus for control room operators should be readily available. Control room floors, ceiling, supporting structures, and walls, including penetrations and doors, should be designed to a minimum fire rating of three hours. All penetration seals should be air tight.

CPNPP Fire Protection Program

Breathing apparatus are located in the Control Room

APCSB 9.5.1 Appendix A

F.2 The control room ventilation intake should be provided with smoke detection capability to automatically alarm locally and isolate the control room ventilation system to protect operators by preventing smoke from entering the control room.

CPNPP Fire Protection Program

The Control Room air intake is provided with smoke detection which alarms in the Control Room so that the ventilation system can be manually placed in the isolation mode upon detection of smoke in the intake duct.

APCSB 9.5-1 Appendix A

F.2 Manually operated venting of the control room should be available so that operators have the option of venting for visibility.

CPNPP Fire Protection Program

The ventilation systems can be manually operated as required for smoke venting. Portable equipment can be used for supplementary venting.

APCSB 9.5-1 Appendix A

F.2 Cables should not be located in concealed floor and ceiling spaces. All cables that enter the control room should terminate in the control room. That is, no cabling should be simply routed through the control room from one area to another.

CPNPP Fire Protection Program

All cables entering the Control Room terminate in the Control Room. There is a small amount of cabling which is enclosed in steel conduit, except telephone cable between junction boxes and handset stations, and a small amount of metal clad cable routed above the Control Room suspended ceiling for which detection is provided.

There is a small amount of metal clad cable or cabling enclosed in steel conduit routed under the raised platform of the Control Room center console. Detection is not provided for this under platform location.

APCSB 9.5-1 Appendix A

F.3 Cable Spreading Room

F.3.a The preferred acceptance methods are:

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline.

APCSB 9.5-1 Appendix A

F.3.a.1 Automatic water system such as closed head sprinklers, open head deluge, or open directional spray nozzles. Deluge and open spray systems should have provisions for manual operation at a remote station; however, there should also be provisions to preclude inadvertent operation. Location of sprinkler heads or spray nozzles should consider cable tray sizing and arrangements to assure adequate water coverage. Cables should be designed to allow wetting down with deluge water without electrical faulting. Open head deluge and open directional spray systems should be zoned so that a single failure will not deprive the entire area of automatic fire suppression capability. The use of foam is acceptable, provided it is of a type capable of being delivered by a sprinkler or deluge system, such as an Aqueous Film Forming Foam (AFFF).

CPNPP Fire Protection Program

Primary fire suppression for each Cable Spreading Room is provided by a total-flooding Halon 1301 system. Class “A” detection loop circuitry is used to actuate the Halon suppression system. A manual pre-action sprinkler system is provided as backup to the halon system.

Cables are designed to allow wetting without electrical faulting.

APCSB 9.5-1 Appendix A

F.3.a.2 Manual hoses and portable extinguishers should be provided as backup.

CPNPP Fire Protection Program

Two hose stations and portable extinguishers are provided for each cable spreading area. See Section 9.5.1.4.2.

APCSB 9.5-1 Appendix A

F.3.a.3 Each cable spreading room of each unit should have divisional cable separation, and be separated from the other and the rest of the plant by a minimum three hour rated fire wall (Refer to NFPA 251 or ASTM E-119 for fire test resistance rating).

CPNPP Fire Protection Program

The unit 1 and 2 cable spreading rooms are separated from each other as well as from the adjacent plant areas by three hour rated fire walls except as noted in Reference [19].

APCSB 9.5-1 Appendix A

F.3.a.4 At least two remote and separate entrances are provided to the room for access by fire brigade personnel; and

CPNPP Fire Protection Program

Two remote and separate entrances from two different fire areas are provided for each cable spreading room.

APCSB 9.5-1 Appendix A

F.3.a.5 Aisle separation provided between tray stacks should be at least three feet wide and eight feet high.

CPNPP Fire Protection Program

Most cabling, except conduit and cable trays run adjacent to the walls, is run to provide a minimum height of eight feet between the lowest tray and the floor. This arrangement allows more free area and is more practical for manual firefighting than aisle separation.

APCSB 9.5-1 Appendix A

F.3.b For cable spreading rooms that do not provide divisional cable separation of F.3.a.3, in addition to meeting F.3.a.1, .2, .4 and .5 above, the following should also be provided:

F.3.b.1 Divisional cable separation should meet the guidelines of Regulatory Guide 1.75, “Physical Independence of Electric Systems.”

CPNPP Fire Protection Program

See Section 8.3.1.4.

APCSB 9.5-1 Appendix A

F.3.b.2 All cabling should be covered with a suitable fire retardant coating.

CPNPP Fire Protection Program

See Section 9.5.1.6.2. Also, see Guideline F.3.b.4.

APCSB 9.5-1 Appendix A

F.3.b.3 As an alternate to F.3.a.1 above, automatically initiated gas systems (Halon or CO_2) may be used for primary fire suppression, provided a fixed water system is used as a backup.

CPNPP Fire Protection Program

See Section F.3.a.1.

APCSB 9.5-1 Appendix A

F.3.b.4 Plants that cannot meet the guideline of Regulatory Guide 1.75, in addition to meeting F.3.a.1, .2, .4 and .5 above, an auxiliary shutdown system with all cabling independent of the cable spreading room should be provided.

CPNPP Fire Protection Program

The plant is capable of being safely shutdown in the event a design basis fire occurs in the cable spreading room. Alternate shutdown systems and procedures are provided using shutdown paths which are independent of the cable spreading room. See Section 7.4 for more discussion.

APCSB 9.5-1 Appendix A

F.4 Plant Computer Room

Safety-related computers should be separated from other areas of the plant by barriers having a minimum three hour fire resistant rating. Automatic fire detection should be provided to alarm and annunciate in the control room and alarm locally. Manual hose stations and portable water and halon fire extinguishers should be provided.

CPNPP Fire Protection Program

The plant computers are not safety-related.

APCSB 9.5-1 Appendix A

F.5 Switchgear Rooms

Switchgear rooms should be separated from the remainder of the plant by minimum three hour rated fire barriers to the extent practicable. Automatic fire detection should alarm and annunciate in the control room and alarm locally. Fire hose stations and portable extinguishers should be readily available.

Acceptable protection for cables that pass through the switchgear room is automatic water or gas agent suppression. Such automatic suppression must consider preventing unacceptable damage to electrical equipment and possible necessary containment of agent following discharge.

CPNPP Fire Protection Program

There are two safety related switchgear rooms in each unit: Train A and Train B. Each switchgear room is in a separate fire area. Manual pre-action water sprinklers are provided for general area protection and/or for direct water spray on cable trays wherever there is a concentration of cable trays. Manual hose stations and portable extinguishers are also provided in these areas.

A fire detection system which alarms at the local detection panel and in the control room is provided for both safety-related switchgear areas. Electrical equipment which could be damaged by direct water spray was considered. See Guideline D.3.c.

APCSB 9.5-1 Appendix A

F.6 Remote Safety Related Panels

The general area housing remote safety related panels should be provided with automatic fire detectors that alarm locally and alarm and annunciate in the control room. Combustible materials should be controlled and limited to those required for operation. Portable extinguishers and manual hose stations should be provided.

CPNPP Fire Protection Program

The general area, housing safety related equipment, is provided with automatic detectors that alarm at the local control panel and annunciate in the Control Room as determined in the Fire Protection Report. Administrative procedures have been developed to control combustible materials. Portable extinguishers and manual hose stations are provided.

APCSB 9.5-1 Appendix A

F.7 Station Battery Rooms

Battery rooms should be protected against fire explosions. Battery rooms should be separated from each other and other areas of the plant by barriers having a minimum fire rating of three hours inclusive of all penetrations and openings. (See NFPA 69, “Standard on Explosion Prevention Systems.”) Ventilation systems in the battery rooms should be capable of maintaining the hydrogen concentration well below 2 percent volume hydrogen concentration.

Standpipe and hose and portable extinguishers should be provided.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. Safety related battery rooms are separated from each other and other areas of the plant by three hour fire barriers. The ventilation system is capable of maintaining the hydrogen concentration well below two percent by volume.

Hose stations and portable extinguishers are located in the corridors serving the battery rooms.

APCSB 9.5-1 Appendix A

F.8 Turbine Lubrication and Control Oil Storage and Use Areas

A blank fire wall having a minimum resistance rating of three hours should separate all areas containing safety related systems and equipment from the turbine oil system.

CPNPP Fire Protection Program

Walls separating safety related areas from the Turbine Building have a minimum fire rating of three hours. All penetrations in these walls also have three hour ratings, except for bus duct penetrations which are discussed in D.1.j of Section 9.5.1.6.2.

APCSB 9.5-1 Appendix A

F.9 Diesel Generator Areas

Diesel generators should be separated from each other and other areas of the plant by fire barriers having a minimum fire resistance rating of three hours.

Automatic fire suppression such as AFFF foam, or sprinklers should be installed to combat any diesel generator or lubricating oil fires. Automatic fire detection should be provided to alarm and annunciate in the control room and alarm locally. Drainage for fire fighting water and means for local manual venting of smoke should be provided.

Day tanks with total capacity up to 1100 gallons are permitted in the diesel generator area under the following conditions:

a. The day tank is located in a separate enclosure, with a minimum fire resistance rating of three hours, including doors or penetrations. These enclosures should be capable of containing the entire contents of the day tanks. The enclosure should be ventilated to avoid accumulation of oil fumes.

b. The enclosure should be protected by automatic fire suppression systems such as AFFF or sprinklers.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline except as noted in Section 9.5.1.6.2. The day tanks are located in separate fire areas located above their respective diesel generator. The day tank area is protected by an automatic preaction sprinkler system actuated by Class "B" detector loop circuitry.

APCSB 9.5-1 Appendix A

F.10 Diesel Fuel Oil Storage Areas

Diesel fuel oil tanks with a capacity greater than 1100 gallons should not be located inside the buildings containing safety related equipment. They should be located at least 50 feet from any building containing safety related equipment, or if located within 50 feet, they should be housed in a separate building with construction having a minimum fire resistance rating of three hours. Buried tanks are considered as meeting the three hour fire resistance requirements. See NFPA 30, "Flammable and Combustible Liquids Code," for additional guidance.

When located in a separate building, the tank should be protected by an automatic fire suppression system such as AFFF or sprinklers.

Tanks, unless buried, should not be located directly above or below safety related systems or equipment regardless of the fire rating of separating floors or ceilings.

CPNPP Fire Protection Program

The diesel fuel oil storage tanks are located outside and adjacent to the Safeguards Buildings. These tanks are buried. The alternate guideline criteria are not applicable to the CPNPP design criteria.

APCSB 9.5-1 Appendix A

F.11 Safety Related Pumps

Pump houses and rooms housing safety related pumps or other safety related equipment should be separated from other areas of the plant by fire barriers having at least three hour ratings. These rooms should be protected by automatic sprinkler protection unless a fire hazards analysis can demonstrate that a fire will not endanger other safety related equipment required for safe plant shutdown. Early warning fire detection should be installed with alarm and annunciation locally and in the control room. Local hose stations and portable extinguishers should also be provided.

Equipment pedestals or curbs and drains should be provided to remove and direct water away from safety related equipment.

Provisions should be made for manual control of the ventilation system to facilitate smoke removal if required for manual fire fighting operation.

CPNPP Fire Protection Program

The Fire protection Program provides protection in accordance with the guideline for fire safe shutdown components, for required separation criteria see D.1.a of Section 9.5.1.6.1.

Fire areas containing required fire safe shutdown pumps are separated from other fire areas of the plant by three hour fire rated barriers.

The Fire Protection Report demonstrates that an adequate level of fire protection is provided to separate redundant equipment used for safe plant shutdown in the event of a fire. Automatic detection is provided in these areas except where a Fire Hazards Analysis Evaluation has concluded that an equivalent level of protection is provided.

APCSB 9.5-1 Appendix A

F.12 New Fuel Area

Hand portable extinguishers should be located within this area. Also, local hose stations should be located outside but within hose reach of this area. Automatic fire detection should alarm and annunciate in the control room and alarm locally. Combustibles should be limited to a minimum in the new fuel area. The storage area should be provided with a drainage system to preclude accumulation of water.

The storage configuration of new fuel should always be so maintained as to preclude criticality for any water density that might occur during fire water application.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. A fire detection system which alarms locally and in the Control Room is provided above the new fuel storage vault. Hose stations and portable extinguishers are provided adjacent to this area. Also see Section 9.1.1.1.

APCSB 9.5-1 Appendix A

F.13 Spent Fuel Pool Area

Protection for the spent fuel pool area should be provided by local hose stations and portable extinguishers. Automatic fire detection should be provided to alarm and annunciate in the control room and to alarm locally.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline.

APCSB 9.5-1 Appendix A

F.14 Radwaste Building

The radwaste building should be separated from other areas of the plant by fire barriers having at least three hour ratings. Automatic sprinklers should be used in all areas where combustible materials are located. Automatic fire detection should be provided to annunciate and alarm in the control room and alarm locally. During a fire, the ventilation systems in these areas should be capable of being isolated. Water should drain to liquid radwaste building sumps.

Acceptable alternative fire protection is automatic fire detection to alarm and annunciate in the control room, in addition to manual hose stations and portable extinguishers consisting of hand held and large wheeled units.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the alternate criteria of the guideline except as noted in Section 9.5.1.6.2. The CPNPP design does not have a Radwaste Building. The radwaste areas are located in the Auxiliary and Fuel Buildings. These areas are protected by automatic fire detectors, hand held portable extinguishers, and manual hose stations.

APCSB 9.5-1 Appendix A

F.15 Decontamination Areas

The decontamination areas should be protected by automatic sprinklers if flammable liquids are stored. Automatic fire detection should be provided to annunciate and alarm in the control room and alarm locally. The ventilation system should be capable of being isolated. Local hose stations and hand portable extinguishers should be provided as backup to the sprinkler system.

CPNPP Fire Protection Program

The decontamination areas are provided with automatic fire detectors, manual hose stations and portable extinguishers. Quantities of flammable liquids stored in these areas will not exceed the amounts used in the determination of the combustible loading.

APCSB 9.5-1 Appendix A

F.16 Safety Related Water Tanks

Storage tanks that supply water for safe shutdown should be protected from the effects of fire. Local hose stations and portable extinguishers should be provided. Portable extinguishers should be located in nearby hose houses. Combustible materials should not be stored next to outdoor tanks. A minimum of 50 feet of separation should be provided between outdoor tanks and combustible materials where feasible.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline.

APCSB 9.5-1 Appendix A

F.17 Cooling Towers

CPNPP Fire Protection Program

The CPNPP design does not incorporate cooling towers.

APCSB 9.5-1 Appendix A

F.18 Miscellaneous Areas

Miscellaneous areas such as records storage areas, shops, warehouses, and auxiliary boiler rooms should be so located that a fire or effects of a fire, including smoke, will not adversely affect any safety related systems or equipment. Fuel oil tanks for auxiliary boilers should be buried or provided with dikes to contain the entire tank contents.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline. All miscellaneous areas are separated from areas containing safety-related equipment. The CPNPP design includes a fuel oil fired auxiliary boiler including associated above ground fuel oil tank which is diked to contain the entire content of the tank.

APCSB 9.5-1 Appendix A

G. Special Protection Guidelines

G.1 Welding and Cutting, Acetylene-Oxygen Fuel Gas Systems

This equipment is used in various areas throughout the plant. Storage locations should be chosen to permit fire protection by automatic sprinkler systems. Local hose stations and portable equipment should be provided as backup. The requirements of NFPA 51 and 51B are applicable to these hazards. A permit system should be required to utilize this equipment.

CPNPP Fire Protection Program

Storage of this equipment is located outside of areas containing safety-related equipment. The use of such equipment is addressed in Section B of this comparison. Additional portable

extinguishers are provided in the areas whenever such equipment is brought into and used in the plant.

APCSB 9.5-1 Appendix A

G.2 Storage Areas for Dry Ion Exchange Resins

Dry ion exchange resins should not be stored near essential safety related systems. Dry unused resins should be protected by automatic wet pipe sprinkler installations. Detection by smoke and heat detectors should alarm and annunciate in the control room and alarm locally. Local hose stations and portable extinguishers should provide backup for these areas. Storage areas of dry resin should have curbs and drains. (Refer to NFPA 92M, “Water-proofing and Draining of Floors.”)

CPNPP Fire Protection Program

The Fire Protection Program complies with this guideline except as noted in section 9.5.1.6.2. Dry ion exchange resins are not stored in or adjacent to areas containing safety-related systems. Storage areas for dry ion exchange resins are protected by fire detectors, portable extinguishers, and manual hose stations.

APCSB 9.5-1 Appendix A

G.3 Hazardous Chemicals

Hazardous chemicals should be stored and protected in accordance with the recommendations of NFPA 49, “Hazardous Chemicals Data.” Chemical storage areas should be well ventilated and protected against flooding conditions since some chemicals may react with water to produce ignition.

CPNPP Fire Protection Program

The Fire Protection Program is in compliance with the guideline except as noted below. Hazardous chemicals are stored in tanks approved for such service and in areas not containing safety-related components and systems, with the exception of the chemical additive tanks in the Unit 1 and Unit 2 Safeguards Buildings. The chemical additive tanks are located in partially enclosed alcoves in fire areas SB4 and 2SB4. These tanks are not open to atmosphere and are inerted with nitrogen. The locations of the tanks are not considered hazardous.

APCSB 9.5-1 Appendix A

G.4 Materials Containing Radioactivity

Materials that collect and contain radioactivity such as spent ion exchange resins, charcoal filters, and HEPA filters should be stored in closed metal tanks or containers that are located in areas free from ignition sources or combustibles. These materials should be protected from exposure to fires in adjacent areas as well. Consideration should be given to requirements for removal of isotopic decay heat from entrained radioactive materials.

CPNPP Fire Protection Program

The fire protection program is in compliance with the guideline

9.5.1.6.2 Justification for Items of Noncompliance to Appendix A to Branch Technical Position APCSB 9.5-1

The following statements are justification for items of noncompliance to Appendix A to Branch Technical Position APCSB 9.5-1 of Standard Review Plan 9.5.1, Revision 1, as stated in the applicable items of Subsection 9.5.1.6.1. Any additional deviations have been included in Appendix C of Reference [19].

Guideline D.1.d

Guideline D.1.d limits the flame spread, smoke and fuel contribution to a maximum of 25 for interior wall and structural components, thermal insulation materials and sound proofing.

Pipe and tank insulating materials as well as finishing cement, etc. are used in various areas of the plant. This insulation is rated as Flame Spread-25, Fuel Contribution-25, and Smoke Development-50.

Carpet, installed in the Control Room envelope complies with Class I interior floor finish requirements of NFPA 101, 1991 Edition. The Control Room envelope is described in Section 6.4.2.

Thermal insulation for ducts have ASTM E 84 rating of 25, 50, 50.

Thermal insulation for chiller unit heat exchangers and piping have ASTM E 84 rating less than flame spread 25, fuel contribution 30 and smoke developed 150.

Guideline D.1.j

Guideline D.1.j address 3-hour rated floors, walls and ceilings separating fire areas. The following justifications are provided where installations are shown to be adequate through analysis in lieu of providing a tested configuration.

1. Floors, walls and ceilings

   Stair tower walls are constructed of two hour rated of design. Justification is provided in subsection 9.5.1.6.2, guideline D.4.f.

   Removable concrete block walls are not fire tested design. Justification is provided in reference [19]. This justification applies to all applications.

   Protection provided by metal hatch covers installed in three (3) hour rated floors has been demonstrated through analysis in lieu of providing a tested configuration. The combustible loading below the hatches is less than 15 minutes with automatic suppression and detection above and below the hatches. The hatches are coated with a layer of fire resistive material to provide a three (3) hour structural steel resistance. Based on the combustible loading, automatic suppression and detection, and fire

resistive coating, a one hour fire could occur without breaching the fire barrier through the metal hatches.

2. Penetration Seals

Containment electrical seals are not a fire tested configuration. Justification is provided in reference [19].

Containment mechanical seals are not a fire tested configuration. Justification is provided in reference [19].

Protection provided by the penetration seals installed in bus duct penetrations installed in three (3) hour rated barriers has been demonstrated through analysis in lieu of providing a tested configuration. The penetration seal design is similar to one currently used in the plant which has a three (3) hour fire rating. The seal maintains the thickness and continuity of the barrier. The barrier's purpose is unchanged by the bus duct penetration. The fire protection features in the vicinity of bus duct penetrations are adequate for the hazards of the area. Based on the fire protection features and a review by a Fire Protection Engineer, bus duct penetrations are expected to survive a fire severity of three (3) hours without breaching the barrier.

3. Non-Rated Fire Doors

Missile resistant doors are not fire tested assemblies. Justification is provided per reference [19].

Watertight doors are not fire tested assemblies. Justification is provided per reference [19] for redundant safe shutdown related separation barriers. The justifications are also applicable for doors in fire barriers that do not separate redundant safe shutdown systems.

Bullet resistant and penetration resistant doors are not fire rated assemblies. The door assemblies are of a construction similar to units tested and listed by Underwriters' Laboratory subsequent to procurement and installation of the CPNPP assemblies except for the Cable Spreading Room (BR/PR) door. This door is justified per reference [19].

Containment Air-Locks for personnel and emergency escape use are not fire tested assemblies. Justification is provided per reference [19].

Protection provided by the tornado vent/fire dampers installed in fire rated barriers with frames mounted outside the concrete walls on steel angles has been demonstrated through analysis in lieu of providing a tested configurations. The support frames of the assemblies are protected with an approved coating to yield a fire resistance equal to that of the barrier.

Fire damper support frames are more substantial than those used in standard sleeve installations. UL 555 gives the acceptance criteria which specifies that a damper assembly must remain in the opening during the fire, and during hose stream application and that no through openings be created. Based on the substantial support frames, the high probability of the dampers remaining in the opening, and the UL test acceptance

criteria, the dampers are expected to provide a tortuous path for fire propagation and meet the conditions of acceptance in a fire test.

Fire door frames are mounted in a frame of steel angles. These angles are then coated with Thermo-lag fire proofing material in accordance with U.L. Design No. X-611. These fire door assemblies are not expected to compromise the integrity of their host 3 hour fire barriers when exposed to a postulated fire.

Guideline D.2.b

Guideline D.2.b addresses bulk gas storage and tank orientation with relations to building walls.

The CPNPP bulk gas storage tanks are located 350 feet from the nearest safety-related building (Electrical and Control Building). The special separation between the tanks and nearest primary plant building is well in excess of the NFPA requirements and therefore the tank orientation is considered acceptable.

Guideline D.3.d:

Protection provided by the penetration seals installed in flexible conduit penetrations installed in fire rated barriers which separate buildings has been demonstrated through analysis in lieu of providing a tested configuration. Flexible conduits are sealed on both sides of a barrier, which is similar to a tested configuration. The combustible loading is low in the areas of flexible conduit penetrations, and fire protection features adequate for the hazards in the area have been provided. Based on the similarity of the configurations to tested configurations, detection, automatic suppression, and manual fire fighting capability, any fire zone which has a flexible conduit could have a fire severity of 3 hours without breaching the barrier through any of the flexible conduit penetrations.

Guideline D.3.f

Guideline D.3.f requires electric cable construction to meet as a minimum the current IEEE-383 Flame Test.

Fire test standards IEEE 1202, UL 1581, UL 1685, ICES T-29-520, CSA FT-4, 1EC60332-3, UL 910, and UL 1666 have been evaluated as equal to or better than the IEEE 383-1974 test for evaluating fire retardant characteristics of electrical cables. Therefore, these standards are considered acceptable alternatives to IEEE 383-1974 for non-class 1E cables.

Additionally, non-class 1E cables are not required to meet IEEE-383-1974 flame retardancy requirements if they are totally contained within conduit, non-class 1E equipment, or flame retardant covering. However, isolated and limited instances exist where exposed cabling has been utilized which does not meet IEEE-383 flame test acceptance criteria nor that associated with the fire test standards listed above. These instances are described and technically justified below on a case-by-case basis.

A small portion of low capacitance non-IEEE-383 cable is installed in the Control Room cable spreading room and computer room for Unit 1. This cable is associated with the ERF and Data Acquisition computer systems. Justification is provided in item 5a of reference [19].

Another small amount of non-IEEE-383 cable is used in association with the radiation monitoring, communication (telephone, radio interconnector and radio antenna), security systems, and fuel handling bridge crane. A small portion of cable associated with the telephone and radio interconnection systems, cable associated with the Unit 2 Secondary Sampling System Oxygen Analyzer. These cables are all routed in conduit (except for short flexible connectors to the oxygen analyzer detectors, the interior radio antenna and telephone cable between junction boxes and handset stations, and are designed for low power service. They do not present a fire hazard in the areas where they are installed.

Another small amount of non-IEEE 383 cable is used in association with the Loose Parts Monitoring System. The cable is constructed of a conductor encased in a stainless steel sleeve. The assembly is then covered with an essentially noncombustible fiberglass wrapping. The instrument cables are installed inside containment and do not present an appreciable fire hazard in the areas where they are located.

A small amount of non-IEEE 383 cable and electrical tape (jacketing and insulation) is installed in the connection boxes for the Unit 1 and Unit 2 Station Service Water Pump Motors, Component Cooling Water Pump Motors, and Emergency Diesel Generators. The cables and tape are part of the Partial Discharge Monitoring System Bus Couplers. These couplers are used to facilitate real time connection of partial discharge monitoring diagnostic equipment. The cables and tape are fully contained inside the motor/generator connection box and do not represent a fire hazard in the areas where they are installed.

Guideline D.4.f

Guideline D.4.f addresses the fire rating of elevator towers and stairwells outside containment. Barriers enclosing the elevatorshafts are rated at two hours with 1 1/2 hour UL labeled fire door assemblies at openings to the elevator shaft. The elevators are not used during fire emergencies. Stairwells used for egress routes have a two-hour fire resistance rated walls with 1-1/2 hour UL labeled fire door assemblies at all openings into the stairwell. Based on the intended use of the stairwells, it is determined that the two-hour rating is adequate. The presence of a stairwell or elevator tower in a fire area will not degrade the 3-hour fire rating of that area's fire barrier. This is based on the tower creating two 2-hour rated fire barriers in series between fire areas joined by the common tower.

Guideline D.5.a

Guideline D.5.a addresses fixed emergency lighting requirements in the event of a fire.

The control room is provided with AC Essential Lighting and DC Emergency Lighting. The AC Essential Lighting is powered from the onsite Standby Diesel Generators. The DC Emergency Lighting is powered from the dedicated non-Class 1E batteries which are sized to supply DC power requirements for a minimum of 8-hours. Fires in areas outside of the control room and cable spreading room will not preclude the availability of DC Emergency Lighting for the control room.

Guideline E.2.e

Guideline E.2.e requires that the water supply be sized assuming all sprinkler heads in the largest designed fire area operating, plus a 1000 GPM hose station allowance. This guideline is

overconservative. The CPNPP water supply is sized following the requirements of NFPA Code 13 with 1000 GPM hose stations allowance.

Guideline E.3.a

Guideline E.3.a requires water flow alarms for standpipe systems. This guideline is not justified because standpipe systems are manual systems requiring plant personnel to be aware of the fire condition prior to the operation of hose stations.

Guideline F.1.a

Guideline F.1.a addresses fixed fire protection for hazards that could jeopardize safe plant shutdown due to a fire inside the Containment Building. An analysis was performed to demonstrate that sufficient equipment is available in at least one shutdown path to safely shutdown CPNPP in the event of an exposure fire in the Containment Building. Radiant Energy Shields were added to resolve interactions as a result of this analysis, and justifications for other interactions present were provided in References [19].

Guideline F.2

Guideline F.2 requires the installation of smoke detectors in the control room cabinets. CPNPP has provided detectors inside control room consoles and equivalent detection for all other control room cabinets.

1. The area immediately adjacent to the cabinets is monitored by ionization type smoke detectors.

2. The Alternate Shutdown System provides safe shutdown of the plant independently of the control room.

3. The area is continuously manned, thus ensuring prompt fire detection.

4. The cabinets were constructed with non-flammable materials containing only low power circuits.

Guideline F.3.b.2

Guideline F.3.b.2 addresses installation of fire retardant coatings on cables in the cable spreading rooms of redundant divisions that are not separated by fire barriers rated at 3 hours fire resistance.

In lieu of installation of 3 hour fire barriers between cable divisions or coating all cables in the area, alternate safe shutdown capability independent of the cable spreading rooms is provided.

Guideline F.9

Guideline F.9 addresses the separation criteria, fire protection criteria, and capability of the diesel generator fuel oil day tanks. The day tanks for each of the four diesel generators are separated and protected in accordance with the criteria of the subject guideline.

The 2160 gallon capacity of the day tank is based on the diesel engine operating characteristic and the operation time. As previously stated, the tank is located in a separate enclosure provided with an automatic preaction sprinkler system. Based on the above and the operation requirements, the existing day tank arrangement is acceptable.

Guideline F.14

Guideline F.14 addresses the use of large wheeled extinguishers. CPNPP has provided hand held extinguishers per NFPA 10 to protect against the potential hazard created by the Radwaste Areas.

Guideline G.2

Guideline G.2 addresses automatic fire protection for areas storing dry ion exchange resins. Storage of dry ion exchange resins is located in the Turbine Buildings remote from areas encompassing safety-related equipment and from any potential ignition sources and exposure to fire hazards. The storage arrangement and the type of resins and quantity stored do not present a fire hazard. Based on the above, protection of the area by manual hose stations and portable extinguishers is adequate.

9.5.2 COMMUNICATION SYSTEMS

9.5.2.1 Design Bases

A comprehensive communications system is provided to ensure reliable intraplant communications, plant to offsite telephone and carrier communications, and offsite emergency communications capabilities with public safety agencies. Effective communication between personnel during plant startup, operation, shutdown, refueling, and maintenance activities is provided by the use of Intraplant (IP) telephone, plant radio, sound-powered telephone, Gai-Tronics page/party, plant-to-offsite two-way radio, wireless intercom system or plant-to-offsite emergency telephone systems. These diverse means of communication are independent to prevent the loss of all systems as a result of a single failure.

An emergency alarm system is installed which provides a unique alarm signal to ensure personnel evacuation.

9.5.2.2 System Description

The following systems comprise the intraplant and plant-to-offsite communication systems for both units.

9.5.2.2.1 Gai-Tronics Page/party System

The Gai-Tronics Page/party System provides party lines (two or more persons), plant wide paging system and plant wide alarms signaling. The page-party line loud speakers are supplied from a source which is available upon Loss of Offsite Power.

The system layout permits communication between the Control Room and all plant areas and buildings of the two units. The system also permits two-way communication between two or more locations. Speakers and microphone handsets are installed at locations vital to the

operation of the plant and the safety of personnel. The voice paging channel output is audible over the expected noise levels under both normal and accident conditions.

Three separate paging zones, consisting of the Control Room Zone, Administrative Buildings Zone, and Main Plant Zone are provided to minimize paging interference from the remaining zones. Manually operated selector switches are provided in the Control Room and other key locations to allow plant wide paging.

Four separate party lines are provided to permit communication between handsets only, thereby making the page channel available to others. All four party lines are available at all handset stations, except those in elevators where only one party line is available. Selection of a desired channel is achieved by a multiposition switch provided as a part of the handset station. Both the page channel and the party line channels may be used simultaneously without interference.

A page-party line (with only one party line) handset station is installed in plant elevators to permit communication in emergency situation.

9.5.2.2.2 Intraplant Telephone System

An independent touchtone telephone system, the IP telephone system is provided for uninterrupted private communication between the following areas: the Control Room, Fuel Building, health/physics and instrument shop areas, hot shutdown panel area, hot shop, Guard House, reactor operating platform areas, intake structures, Maintenance Building offices, and Administration Building offices and work areas. The IP telephone system interfaces with the Intraplant Communication System through an isolating device to ensure that a single failure in either one of these two systems does not affect safe and reliable operation of the other system. Power is supplied to the IP telephone system from the non-Class 1E bus. When the IP telephone system's normal AC power supply is lost, predetermined telephone stations remain operable which derive their power from the public telephone system.

The IP telephone system is connected to the public telephone system by trunk lines as described in Subsection 9.5.2.2.5.

9.5.2.2.3 Intraplant Sound-Powered Telephone System

This system consists of one subsystem per unit as follows:

Subsystem One: Maintenance Loops - Consists of a two channel hard-wired communication link between the control room area and critical plant areas.

The headset jack stations are conveniently located on panels in the Control Room and in critical areas.

Communication can be established between the Control Room and any local panel or between two local panels by suitably plugging the headsets into jack stations which are mounted either in the panel or nearby. This system provides standby communication capability and does not depend on external sources of power other than the human voice.

9.5.2.2.4 Plant Radio Transmitter Receiver System

The plant radio transmitter receiver system provides communication for the fire brigade and the plant operators by the use of two separate repeater systems. Each repeater system receives low power signals from radio units on one frequency and transmits at a higher power level allowing extended radio coverage.

“Satellite Receivers” are strategically placed in areas of the plant to ensure clear reception of low power portable transmitters from at least one of the receivers inside the plant area. All receivers are tuned to the same frequency so that, as the portable radio is moved from place to place throughout the plant, another receiver can pick up the signal. All receiver signals are simultaneously voted by a “comparator” unit to select the receiver with the best quality signal. This signal is then sent to the repeater system for retransmission to all monitoring radio units.

Also used to improve reception is “radiax” transmission cable. The radiax transmission line, which is located throughout the plant, leaks a controlled amount of RF (radio frequency) energy over its full length allowing radio communication to be maintained for areas in which radio attenuation of RF signals, due to extensive metal and concrete building construction, makes normal propagation from a single antenna unsuitable.

9.5.2.2.5 Public Telephone System

The public telephone system is interconnected to the Intraplant (IP) Telephone System by trunk lines. This permits access to the public telephone system from the Control Room, health/physics and instrument shop areas, hot shutdown panel, Guard House, Hot Shop, Maintenance Building offices, Administration Building offices and work areas, reactor operating platform area, and intake structures.

9.5.2.2.6 Plant-to-Offsite Two-Way Radio Transmitter-Receiver System

A VHF two-way radio transmitter-receiver system is provided for emergency communication between the plant and offsite public safety agencies.

The system description is provided in the security plan.

9.5.2.2.7 Plant-To-Offsite Emergency Telephone Systems

The emergency telephone system consists of several dedicated leased telephone lines.

1. Emergency Notification System (ENS) - Dedicated telephone line between CPNPP and the NRC Incident Response Center in Bethesda, Maryland to inform the NRC of an emergency.

2. State and County Notification System - Dedicated telephone lines between CPNPP and the designated state office as specified in the Emergency Plan and the local emergency operations centers in Glen Rose and Granbury to inform state and local officials of an emergency.

3. System Dispatch Center Line - A dedicated telephone line from CPNPP to the offsite system dispatching center.

4. Direct Offsite Telephone Lines - Direct telephone lines to offsite that are independent of the IP Telephone System in the event the IP Telephone System fails.

9.5.2.2.8 Wireless Intercom System

The wireless intercom system provides communications for the fuel handling area and reactor operating floor during refueling operations in modes 5 and 6. Communications to the control room from the fuel handling area and reactor operating floor is provided by Gai-Tronics or Plant Radios.

9.5.2.2.9 Emergency Evacuation Alarm System

The evacuation alarm is generated by a solid state multi-tone generator capable of producing five distinctive tones which can be heard over all plant paging zones via the Intraplant Communication System except overhead and wallmounted speakers in the Control Room. One of the distinctive tones, which satisfies the NRC Regulatory Guide 8.5 [3] requirements, is designated for the evacuation alarm signal. The evacuation alarm system includes rotating beam lights as a visual indication for an alarm condition in those high background noise areas where the evacuation alarm is inaudible.

The evacuation alarm system, including the multi-tone generator, is powered by a source available upon loss of offsite power to ensure personnel evacuation in case of an emergency. The alarm is initiated by the Control Room operator in the event of a site evacuation emergency.

9.5.2.2.10 Evaluation

The following evaluation is intended to establish the adequacy and redundancy of the plant communication system design:

1. Intraplant Systems

   Each intraplant system, i.e., Gai-Tronics page/party system, IP telephone system, wireless intercom system, sound-powered telephone system, and plant radio transmitter-receiver system, is designed to provide the required intraplant communications during and after accident conditions as well as for plant operation and maintenance purposes. Failure of any one of the above systems does not result in a failure of any other system. The power supply for the IP telephone system is provided from the non-Class 1E bus. Upon loss of total AC power to the IP telephone system, predetermined telephones remain operable deriving power from the public telephone system.

   The power supply for the intraplant communication system (Gai-Tronics page/party) is provided from a source available upon Loss of Offsite Power. The Gai-Tronics page/party system handsets and speakers are strategically located to cover critical areas. Each area of the plant is served by a separate circuit to confine a system outage to the area served by the faulty circuit. The Gai-Tronics page/party communication system is employed as the primary backup to the plant radio system for use by plant operators during hot shutdown.

The IP telephone and Gai-Tronics page/party systems are connected through an amplifier device which acts as an isolating device. This ensures that a single failure in either of these two systems does not result in a failure of the other system.

The sound-powered telephone system is independent of all external power sources and its headset jack stations are conveniently located throughout the plant. This system can be employed as a backup to the intraplant communication system in critical equipment areas of the plant. In addition, two channels of components and cabling exist in the sound-powered telephone system to prevent failure of both systems because of a single component failure. Reliable service can be expected because of the ruggedness of these components and their independence from any external power source.

Each intraplant radio transmitter-receiver system is provided with DC battery backup. This system is employed as the primary communication system by plant operators and the fire brigade in case of emergency.

Power to the wireless intercom system Master Controller is provided by plant support power and each headset has an individual battery supply. Upon loss of power, backup communications is provided by the Intraplant Radios.

One of the following communication systems is available between the Hot Shutdown Panel and other remote panels required for fire the safe shutdown of the plant:

a. Intraplant Radio Transmitters-Receivers

b. Gai-Tronics Page/Party handset stations and speakers.

2. Plant-to-Offsite Systems

There are three independent plant-to-offsite communication systems available for the use of Control Room operators. The availability of these systems during and after the accident condition is enhanced by the fact that each enters the plant via different means.

The public telephone lines (trunk lines) are connected to the plant IP telephone system. This extends the use of public telephone lines throughout the plant.

The plant-to-offsite two-way radio communication system can be used as a backup to the public telephone system. This provides communication between the plant and public safety agencies. The power to this system is provided from a source available upon loss of offsite power. Also, there is an inherent redundancy in this system since it has two separate independent onsite base stations.

The dedicated telephone line to the system dispatcher provides further redundancy by establishing communication outside the plant area during the plant emergency and the loss of the other two plant-to-offsite communication systems.

9.5.2.3 Inspection and Testing Requirements

All communication systems are inspected and tested (and adjusted if required) at the completion of the installation to ensure proper coverage and audibility under the maximum plant noise levels during the various operating condition, including the accident condition.

Since the communication systems are used on a daily basis, periodic testing is not required.

Periodic testing of the emergency evacuation alarm signal is performed as outlined in ANSI N2.3-1967 [4].

9.5.3 LIGHTING SYSTEMS

9.5.3.1 Normal Lighting System

Normal lighting for the plant is supplied by a grounded 208/120-V, three-phase, four-wire distribution system. Dry-type transformers, which feed lighting panels, are rated at 30 kVA or 45 kVA, 480- 208/120-V, three phase, 60 Hz, delta-wye, and are connected to motor control centers throughout the plant. The transformers and lighting panels which they feed are conveniently located plantwide to permit efficient distribution of the lighting load. Some of the motor control centers are common to both units. Exterior, nonsecurity lighting uses the 208/120-V system described above as well as a 480-V system fed from lighting panels.

The lighting system design is equal to or exceeds the recommendations contained in the Illuminating Engineering Society (IES) Lighting Handbook [17].

9.5.3.2 AC Essential and DC Emergency Lighting Systems

9.5.3.2.1 System Description

AC Essential and/or DC Emergency Lighting systems are provided in those locations where safety-related functions are performed or where personnel safety is involved. They are designed to provide adequate illumination for the safe shutdown of the plant and the evacuation of personnel. These areas include the following:

1. Control Room

2. Diesel generator room

3. Remote shutdown panel locations (which include the Hot Shutdown Panel and the Shutdown Transfer Panel)

4. Areas required for control of safety-related equipment to accomplish alternate shutdown from outside the control room

5. Primary interior access routes to and from the preceding areas

6. Primary exits

The primary plant AC Essential Lighting is provided by AC systems having connections to the onsite standby diesel generators. During all operating conditions or a LOCA and in the event of the loss of all offsite AC power, onsite standby diesel generator power is available to the AC Essential Lighting System. The illumination level provided by the AC Essential Lighting System is as follows:

1. For the Control Room, in the control board area, there is full illumination, which conforms to the requirements of NUREG-0700 [42].

2. For other primary plant areas requiring AC essential lighting, the illumination level ranges from full illumination to approximately one-fifth to one-tenth of full illumination, which conforms to the requirements of Illumination Engineering Society (IES) Handbook [17].

3. Emergency illumination levels provided by AC essential lighting for personnel safety, evacuation, and operation of safe shutdown work stations will meet or exceed the requirements as described under DC Emergency Lighting System in this section.

Outside of the Containment and Fuel Building, AC Essential Lighting is provided in primary plant areas required for safe shutdown and in major interior access/egress routes between these areas. Inside the Containment, AC lighting is provided for emergency egress. This AC lighting is powered from the Class 1E 480V system, which is powered by the standby diesel generators during a loss of offsite power. AC Essential Lighting is not provided in the Fuel Building, Service Water Intake Structure or Turbine Building.

The non-Class 1E DC Emergency Lighting System consists of lights connected to the dedicated batteries or individual battery packs. DC Emergency Lighting is provided in all areas needed for the operation of fire safe shutdown equipment necessary to achieve hot standby and in the primary interior access/egress routes between these areas. This lighting is provided by 8-hour rated battery packs except in the control room where the lighting is provided by dedicated 8-hour batteries. Battery pack lights are flourescent or sealed beam type.

Emergency Lighting is also provided for evacuation paths in the plant.

The DC Emergency Lights in the Control Room are normally deenergized. The contactor in the DC Emergency Lighting panels is normally held open by a feed from the AC Lighting System. DC Emergency Lights are activated by the loss of power to the AC Lighting Systems.

Individual battery packs are normally under a float charge from the AC lighting power supplies for their respective areas. If an AC lighting power supply is lost, the respective DC lights are activated from their individual battery packs by their respective relays.

The lighting (AC and DC) in the Control Room, the primary plant (non-containment) essential control areas and the primary interior access routes between them is arranged in a staggered pattern and alternately fed from the redundant trains and/or from self contained battery packs energized from either Train A, Train B or Normal AC sources. As an alternate, where the engineered safety features (ESF) equipment is energized from a particular train, AC essential and DC emergency lighting from the same train is provided in that area and in the primary access route to it. These practices ensure adequate lighting in these areas under any possible electrical single-failure condition.

The DC Emergency Lighting System provides the following illuminance levels:

1. 0.5 footcandles (fc) average maintained, horizontally, along the center line of fire protection access/egress routes at floor level, with a maximum/minimum uniformity ratio less than 40:1;

2. 0.5 fc minimum maintained at the center point of a slight hazard within the fire protection access/egress route, with 3 fc average maintained along the hazard. Slight hazards are defined, for safety lighting purposes only, as abrupt changes in direction, intersections of two or more corridors or access/egress routes, changes of floor level, at each door, along stairs, and at weather-stripping door sills;

3. 2 fc minimum maintained at the center point of a high hazard within the fire protection access/egress route. High hazards are defined, for safety lighting purposes only, as permanent stationary protrusions (such as piping, cable trays, panelboards, and water-stopping door sills) into a 2ft-6in wide by 6ft-6in high access/egress route envelope. Hazards are to be determined by a field walkdown. Door sill lighting is in horizontal fc; all other hazard lighting is in vertical fc;

4. 10 fc minimum average maintained within the task seeing area of the following work stations:

   Main Control Boards in the Main Control Room

   Hot Shutdown Panels

   Shutdown Transfer Panels

5. The emergency DC lighting illuminance levels at certain additional task areas where specific event-related tasks may be required to be performed for fire protection purposes will be as follows:

   a. 1 fc average maintained within the task seeing area of valves, individual control switches, and other local panel boards required to be illuminated for fire safe shutdown.

   b. 2 fc average maintained within the task seeing area of trip circuits required for fire safe shutdown.

   c. In addition, supplementary lighting will be provided from battery-powered hand held portable lanterns.

9.5.3.2.2 Design Criteria

The non-Class 1E AC Essential Lighting Systems are connected to Class 1E power system buses through Class 1E circuit breakers in 480 volt Motor Control Centers, Class 1E 480 volt-120/208 volt step-down transformers, and Class 1E lighting distribution panels. The interconnection of the non-Class 1E AC essential lighting circuits with the Class 1E power system is discussed in paragraph 7 of Section 8.3.1.2.1 and Appendix 1A(B) of Section 1.0. The power cables from the 480 volt Motor Control Center to the primary side of the 480 volt

step-down transformer and from the secondary side of the step-down transformer to the lighting distribution panel are Class 1E. The cable raceway system for these power cables is designated Seismic Category I. The feeder cables from the lighting distribution panels to the lighting fixture outlet boxes are non-Class 1E.

The AC essential lighting fixtures, fixture primary supports, lamps and bulbs are classified non-nuclear safety grade (NNS).

The DC emergency lighting fixtures, lamps, fixture primary supports, fluorescent and sealed beam lamps and supports, individual sealed beam battery packs and supports are non-nuclear safety grade (NNS) components and material.

Restraints are placed on lighting system fixtures where required in areas of the primary plant to ensure the integrity of safety related equipment during a seismic event.

In the areas in which the use of mercury is restricted, regular or tungsten halogen incandescent lamps are used for the Normal, AC Essential, and DC Emergency Lighting systems. The lighting for security areas is discussed in the security plans.

9.5.3.2.3 Failure Analysis

The AC Essential and DC Emergency Lighting systems are independent of each other. Any electrical failure in one system does not cause the other system to be inoperable. Furthermore, in critical control areas the lighting is either redundant (i.e., overlapping from adjacent areas) or energized from the same train as the equipment in that area and/or provided by self contained battery packs energized from either Train A, Train B or Normal AC sources. The integrated design of these systems provides adequate emergency station lighting in all areas required for control of safety-related equipment and in the major interior access routes to and from these areas.

9.5.3.2.4 Inspections and Testing

The Normal, AC Essential, and DC Emergency Lighting systems are tested and inspected after their installation is complete.

The AC essential lighting is used on a day-to-day basis to provide a part of the ordinary operational lighting. Therefore, periodic tests are not required. Prompt replacement of burned out lamps assures that the AC essential lighting system is capable of performing at light levels well above the minimum requirements.

The DC emergency lights connected to the dedicated batteries can be tested periodically by tripping the circuits fed from the AC lighting system, thereby closing the feeder circuits to the DC emergency lights. For battery packs, the test circuit contained in each individual unit can be used for test purposes. Battery pack batteries are replaced on a periodic basis.

9.5.3.3 Underwater Lighting

High Pressure Sodium underwater lights are used in the spent fuel pools, transfer canals and wet cask loading pit in the Fuel Building. These lights are subject to inspections and precautions and limitations to control the free mercury contained in the bulbs. These controls minimize the

likelihood of the loss of the mercury and requires an evaluation if mercury is lost and not subsequently recovered.

9.5.4 DIESEL GENERATOR FUEL-OIL STORAGE AND TRANSFER SYSTEM

9.5.4.1 Design Bases

The Diesel Generator Fuel-Oil Storage and Transfer System design is in accordance with the following criteria:

1. 10 CFR Part 50, Appendix A, Criterion 2, Design Bases for Protection Against Natural Phenomena

2. 10 CFR Part 50, Appendix A, Criterion 4, Environmental and Missile Design Bases

3. 10 CFR Part 50, Appendix A, Criterion 5, Sharing of Systems, Structures, and Components

4. NRC Regulatory Guide 1.29, Seismic Design Classification

5. NRC Regulatory Guide 1.137, Fuel-Oil Systems for Standby Diesel Generators (1/1978). The CPNPP design conforms to the requirements of this guide with exceptions as described in Section 1A(B).

The criteria for protection against pipe break outside the Containment conforms to the guidelines contained in Branch Technical Positions APCSB 3-1 and MEB 3-1 as described in Section 3.6. The system is designed to meet the single failure criterion. To further its continued availability, the system is designed to withstand the effects of the worst anticipated environmental phenomenon and to meet seismic Category I requirements.

The Diesel Generator Fuel Oil Storage and Transfer System, shown on Figure 9.5-52, contains Nuclear Safety Related portions designed to IEEE 387 and seismic Category I requirements. These portions are designated by a special symbol and accompanying note on the drawing. They have no other significance than this. The nuclear safety class 3 (ANS safety class 3) portion of the system is also designed to the seismic category I requirements (see Table 3.3-2 for safety classification and Section 3.2.1.2 for seismic category of mechanical systems and components). Symbols used for identification of the ANS nuclear safety class portion from that of the ANS non-nuclear safety class are shown on Figure 3.2-1.

The system is designed in accordance with applicable codes as stated in Section 3.2. The system is designed to supply a reliable source of fuel oil for the four emergency diesel generator sets for a period of not less than seven days. The seven day fuel capacity is based on the following:

1. Continuous operation at rated load, as stated in Section 8.3

2. Diesel fuel lower heating value

3. Fuel consumption based on test data

9.5.4.2 System Description

9.5.4.2.1 General

The four diesel generator sets are supplied by four separate, identical, fuel-oil system trains, as shown on Figure 9.5-52. Each train consists of an underground storage tank, two transfer pumps, day tank, piping, instrumentation and controls necessary for reliable operation. Connections are provided for filling and venting the storage tanks, as shown on Figure 9.5-51. Provisions are made for draining and sampling of fuel, and removing condensate from the storage tanks. Each day and storage tank is provided with instrumentation to monitor and alarm the low & high levels of fuel oil. Hi and lo levels are annunciated on the engine control panel and alarm as “Diesel Engine Trouble” in the Control Room.

Two fuel-oil transfer pumps are located in each diesel generator room, as shown on Figure 9.5-52. Power for these pumps is supplied from their respective emergency diesel generator buses, as described in Section 8.3.1. Each pump takes suction from its associated storage tank and discharges to its associated day tank. The general arrangement of the diesel generator area is shown on Figure 9.5-53.

Fuel consumption may be monitored as a function of depth of fuel in the fuel oil day tank. The flow rate of the fuel for the emergency diesel generator fuel oil system may be measured indirectly by level instrumentation.

The diesel fuel oil storage tank is equipped with a vent line with flame arrestor, a level transmitter, a fill line with shut-off valve (as described below) and the capability for use of a stick gauge to measure the fuel level.

The storage tank is filled by truck through a connection located inside a watertight valve box or via an alternate fill method which utilizes a special diffuser tool to fill through the Sample Grab Point quick-disconnect fitting connection, which is also in a watertight box. This prevents dirt and water from mixing with the fuel-oil. In an effort to minimize the creation of turbulence which will result in the stirring of sediment while the Diesel Fuel Oil Storage Tank is being filled, the fill nozzle is equipped with a fill line which is run to within 2 feet of the bottom of the tank, the end of which is capped and the last two (2) feet of which is perforated with approximately sixteen 1" diameter holes. When the alternate fill method is used, the special diffuser tool is also capped at the bottom and perforated with holes. The function of this arrangement of the fill line is to prevent impingement of the new fuel oil on the bottom and to disperse the new fuel oil throughout the fuel oil that is in the tank. A dirt and water collector is provided in order to limit the build-up of sediment and water in the storage tanks. These collectors will be cleaned on an as needed basis.

The growth of algae is not considered a design basis for the diesel generator fuel oil storage and transfer system but may occur at CPNPP. For this reason, a biocide is added to new fuel in order to deter the growth of algae in the fuel. Diesel fuel oil is sampled, tested and maintained in accordance with CPNPP Technical Specifications.

The tank is assumed to contain a low amount of diesel fuel in it during possible flood conditions to provide the most conservative design case. The tank is provided with hold down straps embedded in a concrete foundation. The mass of the concrete foundation counteracts the buoyancy effects and protects against the possibility of tank floatation.

A strainer is located downstream of each redundant fuel oil transfer pump. In addition, duplex strainers are located downstream of the fuel oil day tank on the engine skid. These strainers are provided to catch any suspended sediment.

A 3 to 5 micron duplex filter is located on the engine skid upstream of the fuel supply header as a final measure to collect suspended sediment.

The day tanks supply an immediate source of fuel-oil to the diesel generator sets. Fill, vent, drain connections and a return line to the storage tank for overflow are provided for each tank. In addition, provisions are made for direct filling of the day tanks from tank trucks. Each day tank is provided with level instrumentation which controls fuel-oil transfer pump operation, provides level indication on the engine control panel and initiates low- and high-level alarms in the diesel generator room, which alarm as “diesel engine trouble” in the Control Room.

If new fuel oil does not meet the diesel generator manufacturer’s requirements for absolute specific gravity at 60/60^{o}F of $\geq$ 0.8299 or for API gravity at 60^{o}F of $\leq$ 39^{o}, it is acceptable to add new fuel oil to the storage tank(s) only if, after being added, the entire storage tank(s) will meet the manufacturer’s recommendations.

The fuel oil transfer and storage system utilizes pressure indications on the discharge lines of the fuel-oil transfer pumps and differential pressure alarms on the strainers, located in the discharge lines (see Figure 9.5-52 sheets 1 and 2). Temperature indication is not provided on the fuel-oil transfer system since the maximum anticipated temperature differential for the underground tanks and piping, is not expected to exceed the diesel generator manufacturers recommendations. The instrumentation described above is utilized during all modes of operation.

9.5.4.2.2 Equipment Design Bases

The following are the equipment design bases for the major components of the Diesel Generator Fuel-Oil Storage and Transfer System:

1. Fuel-Oil Storage Tank

   The combined capacity of the fuel-oil storage and day tanks for each diesel generator set is designed for a seven-day fuel supply at rated load (see Section 9.5.4.1 for basis) plus approximate 15 percent margin to permit periodic testing.

2. Fuel-Oil Day Tank

   The fuel-oil day tank is designed to supply enough fuel for a minimum of 1 hour continuous operation at rated load plus a minimum margin of 10 percent as required by ANSI N195.

   The fuel-oil day tank is located in the diesel generator room and is enclosed by three-hr fire walls. This tank is elevated above the diesel generator to maintain a positive pressure at the suction of the engine fuel pumps.

3. Fuel-Oil Transfer Pump

The fuel-oil transfer pump has a capacity equal to 125 percent of the engine's full load fuel requirements.

4. Diesel Generator Sets

Two identical diesel generator sets of equal capacity, characteristics, and interchangeable components are provided for each unit. The diesel generator sets are qualified in accordance with IEEE Standard 387-1977 [16].

The engine is diesel oil, four-cycle, 450 rpm, turbocharged, water-to-water cooled with station service water. The generator is a synchronous machine type, self-ventilated, air-cooled unit with a static exciter, a voltage rating of 6900, and a power factor of 0.8; it is directly connected to its driver.

There is no AC power required to start the diesel generators. The DC power required to start the diesel generators is supplied from Class 1E 125-VDC buses described in Section 8.3.2.

5. Fuel-Oil

The fuel is diesel fuel oil No. 2 in accordance with ASTM D 975-1981 [44] and Technical Specification requirements.

If new fuel oil does not meet the diesel generator manufacturer's requirements for absolute specific gravity at 60/60^{o}F of $\geq$ 0.8299 or for API gravity at 60^{o}F of $\leq$ 39^{o}, it is acceptable to add new fuel oil to the storage tank(s) only if, after being added, the entire storage tank(s) will meet the manufacturer's recommendations.

6. Piping and Valves

The Diesel Generator Fuel-Oil Storage and Transfer System piping & valves are fabricated from either carbon or stainless steel.

9.5.4.3 Safety Evaluation

Safety classification of the system components meets the intent of ANSI N18,2a [15] as endorsed by RG1.137 and ANSI N195. Safety Class 3 applies to components of the Diesel Generator Fuel-Oil Storage and Transfer System as shown on Figure 9.5-52.

Each diesel generator set is provided with a usable storage capacity for seven days of operation (see Section 9.5.4.1 for basis), in accordance with Regulatory Guide 1.137. Approximate 15 percent margin in storage tank capacity is also provided to preclude the necessity of refilling the tanks after routine testing operations.

To ensure the continued integrity of the system, the system components are designed to seismic Category I requirements. See Section 3.7 for details. In addition, the system is protected from tornado-generated missiles. Underground components are protected from corrosion by a coating and the Diesel Fuel Oil Storage Tank by a cathodic protection system.

The components in the system are designed to the requirements of the ASME B&PV Code, Section III, Class 3. However, when an ASME Class 3 design component is commercially unavailable, the component is proven to be of equivalent quality.

“Equivalent quality”, of a component is interpreted to mean an item designed for normal commercial use upgraded to ASME B&PV Code Section III requirements through seismic design, testing, qualification and documentation.

Component supports are designed to ASME B&PV Code, Section III Class 3. Fabrication is as a minimum in accordance with AISC-1970.

All such components and component supports are listed in Table 17A-1.

Plant layout precludes routing of any high energy lines in the Diesel Generator rooms containing the following systems:

The diesel generator fuel oil system, the diesel generator cooling water system, the diesel generator starting air system, the diesel generator lubrication system and the diesel generator combustion and exhaust system.

Except for fire protection piping, all the piping in each diesel generator compartment is associated with a diesel generator of the same train affiliation. The diesel generator compartments are equipped with watertight doors to protect the electrical equipment area and the flooding resulting from a failure of a moderate energy line that is part of the diesel generator or fire protection systems will be limited to the respective diesel generator and not affect the reliability of the redundant diesel generator system. Each diesel generator compartment is connected to the pipe tunnel shown on Figures 1.2-11 and 1.2-17. The penetrations are sealed with fire barriers which also provide a ventilation system boundary. These penetration seals are not required to be watertight since any flooding would be detected by the safeguards building sumps and isolated before it could affect any other equipment in the safeguards building. Therefore, the requirements of BTP APCSB 3-1 as described in Section 3.6B are met. There is no credit taken for the normal drains for the diesel generator shown on Figure 9.3-6. The floor drains for each compartment drain to its own sump; so, backflooding is precluded.

To minimize the chances of fire in the Diesel Fuel Oil System, specific precautions were taken. All fuel oil lines are routed such that they are remote from lines of elevated temperature which are in the same room. Each Fuel Oil Day tank is enclosed in its own cubicle which is rated for three hours of fire separation and is protected by an automatic preaction sprinkler system. The redundant Diesel Generators are separated from each other by three hour rated fire barriers. Fire fighting actions taken for one diesel generator will not affect the redundant diesel generator.

There are no sources of open flame in these areas.

Failures caused by icing or freezing of the outdoor equipment in the Fuel Transfer System are not considered possible because of the low freezing point of diesel fuel.

A fuel-oil transfer pump is started when the day tank level reaches two thirds capacity, which is the low-level set point, and run until the day tank is filled to high level. In the event of pump failure, the two-thirds minimum capacity allows the operator to take corrective action before the diesel generator set runs out of fuel.

The time interval between the low level alarm and when the day tank becomes empty is three hours. The range of malfunctions and necessary remedial actions are considered in Table 9.5-11. Considerable flexibility is provided for resupplying the storage and day tanks. Fuel-oil is normally brought in by tank truck. However, if circumstances require, railroad tank cars can be brought in on the plant railroad spur. CPNPP is located approximately 90 miles southwest of the Dallas - Ft. Worth area. Dallas - Ft. Worth is a major commercial area which has distributors of diesel fuel that represent the majority of the major oil companies. A listing of the cities capable of supplying diesel fuel with their distance in miles is provided below:

City	Area	Distance from Site In Miles
Ft. Worth, Texas	North Texas	50
Dallas, Texas	North Texas	90
Houston, Texas	Texas Gulf Coast	275
Beaumont, Texas	Texas Gulf Coast	300
San Antonio, Texas	South Central Texas	200
Oklahoma City, Oklahoma	Central Oklahoma	250
Abilene, Texas	West Texas	150
San Angelo, Texas	West Texas	175

The failure analysis of the Diesel Generator Fuel Oil Storage and Transfer System is provided in Table 9.5-11.

A failure of the drip waste return pump has no effect on reliable operation of the diesel generator and, consequently, no impact on safe shutdown of the plant. Thus, the component is not required to be active.

9.5.4.4 Inspection and Testing Requirements

Prior to plant initial operation, the diesel generators are installed and thoroughly tested to demonstrate their ability to perform as designed. During these tests all associated equipment are operated. During routine plant operations, the diesel generators are periodically test run at calculated maximum load to ensure their continued capability. At the same time, associated systems are thoroughly inspected and tested where possible. This test run of the diesel generator assures that the fuel-oil is being supplied and is of sufficient quality to allow the diesel generators to perform adequately.

In addition, a comprehensive sampling and testing program is performed on the diesel fuel oil for the emergency generators to assure the quality of fuel oil. The methodology used in sampling diesel fuel follows that outlined in Reference ASTM D4057-1981 [45]. Fuel oil not meeting the requirements of Reference ASTM D2276-1978 [48] is replaced or corrected in a short period of time (about a week).

If new fuel oil does not meet the diesel generator manufacturer's requirements for absolute specific gravity at 60/60°F of ≥ 0.8299 or for API gravity at 60°F of ≤ 39°, it is acceptable to add new fuel oil to the storage tank(s) only if, after being added, the entire storage tank(s) will meet the manufacturer's recommendations.

Fuel oil is sampled before it is transferred to the supply tank. All tests are completed within 31 days of the transfer. At least once every 31 days, the diesel oil is sampled and tested for water and sediment, and the supply tank is checked for condensate which is then removed as necessary. The latter surveillance is increased if condensate levels indicate the need to do so. Day tanks are checked monthly, as a minimum, and after each operation of the diesel where the period of operation is one hour or longer. Any accumulated water is removed immediately. If it is suspected that water has entered the suction piping from the day tank, the entire fuel oil system between the day tank and the injectors is flushed. As a minimum, the fuel oil stored in the supply tanks is removed, the accumulated sediment removed, and the tanks cleaned so that the ASME Section XI, Article IWD-2000, "Examination Requirement" may be performed at the required 10-year interval. Records are kept of all sampling and testing of the diesel fuel oil.

A cathodic protection surveillance is also conducted as follows:

1. At intervals not exceeding 12 months, tests are conducted on each underground cathodic protection system to determine whether the protection is adequate.

2. The test leads required for cathodic protection are maintained in such a condition that electrical measurements can be obtained to ensure the system is adequately protected.

3. At intervals not exceeding two months, each of the cathodic protection rectifiers are inspected.

4. Records of each inspection and test are maintained over the life of the facility to assist in the evaluation of the extent of degradation of the corrosion protection systems.

9.5.5 DIESEL GENERATOR COOLING WATER SYSTEM

9.5.5.1 Design Bases

The Diesel Generator Cooling Water System is designed to allow the diesel generator sets to be rapidly loaded and to operate continuously at their maximum ratings. The various components are sized to remove the maximum heat produced by the diesel generator sets using approximately 117°F service water as a cooling medium. Essential system components are designed to seismic Category I requirements and to withstand the worst anticipated environmental phenomenon as described in Section 3.3, 3.4, and 3.5. This system is designed to meet the requirements of 10 CFR Part 50, GDC 2, 4, 5, 44, 45, and 46, NRC Regulatory Guides 1.29, IEEE 387-1977, and is in accordance with Branch Technical Positions APCSB 3-1 and MEB 3-1 [5 through 14, and 16]. Refer to Section 8.3.1, AC Power System (Onsite), for design information and analyses of standby power systems (diesel generators).

9.5.5.2 System Description

Each Diesel Generator Cooling Water System consists of two subsystems: a closed cooling water loop (jacket water) for the cylinder jackets, cylinder heads, turbochargers, air intercoolers, and lube oil cooler; and an external cooling water loop (service water) which cools the jacket water heat exchanger. (See Figure 9.5-54.) The components for the internal cooling system are tabulated in Table 9.5-12. For design parameters of the jacket water cooler see Table 9.5-18. Operation of the diesel generator for 60 sec without cooling water has no effect on diesel generator performance. If a loss of power occurs, the service water pumps restart within 35 sec after diesel generator power is made available, well within the limits of the Diesel Generator Cooling Water System requirements.

When the diesel generator sets are not operating, a thermostatically controlled electric standby heater and motor-driven keep-warm circulating water pump are provided to reduce the effects of fast start thermal transients on long term maintenance requirements.

The diesel generator sets are capable of operating at rated speed for seven (7) days with no load across the terminals and are capable of assuming design sequenced and continuous loading or any part thereof, during this period. The diesel generator sets have no minimum loading requirements.

The Diesel Generator Cooling Water System is designed as a closed system with a vented standpipe at the highpoint.

The engine driven jacket water pumps required NPSH is 15 feet. The stand pipe provides a static head range of 33 to 44 feet absolute. The stand pipe is also equipped with a low level alarm which is set approximately 2.5 feet below the water level during normal operation.

Assuming that all the water between the alarm set level down to the minimum water level is available for system make-up, then there is potentially 310 gallons of water available to replace a leakage up to 1.5 gallons per hour for seven (7) days of continuous operation.

The Jacket Water Cooling System is supplied with the following instrumentation and alarms for local monitoring of operational conditions of the cooling system and provide indications of any problem that develops within the system:

1. Low-level jacket water standpipe

2. Low-temperature jacket water in

3. Low-temperature jacket water out

4. High-temperature jacket water in

5. High-temperature jacket water out

6. High-high temperature jacket water out trip; the jacket water trips are blocked during safety injection actuation.

7. Low-jacket water pressure

8. Jacket water pressure local indication

9. On-off temperature control for the jacket water heater

Jacket water temperature is maintained constant by the use of an automatic three-way valve which is thermostatically operated. Vents at high points in the system ensure that no air is trapped in the system.

Leakage in the Jacket Water Cooling System can result in a loss of coolant inventory. This will cause a high jacket water temperature alarm followed by a possible trip of running the EDG in NORMAL mode.

Long-term corrosion and organic fouling in the diesel generator cooling water system is minimized by the use of demineralized water mixed with NaNO2.

To prevent degradation of system cooling performance and of the materials utilized, the jacket water heat exchanger is periodically cleaned and flushed during the planned maintenance.

9.5.5.3 Safety Evaluation

Safety classification of the system meets the intent of ANSI N18.2 and N18.2a [15] as shown in Figure 9.5-54.

The system provides an adequate and reliable source of cooling water. The Jacket Water Cooling System and associated service water header for each diesel generator set are totally independent of the others.

The components in the system that are mounted on the engine are constructed in accordance with manufacturers standards to meet performance requirements as stated in IEEE 387, and those which are mounted off the engine are designed in accordance with the ASME B&PV Code, Section III, Class 3. The off engine piping supports are designed in accordance with ASME B&PV Code, Section III, Class 3 and as a minimum fabricated to American Institute of Steel Construction (AISC).

A general diesel trouble alarm which includes pressure, temperature, and coolant system level alarms (which are annunciated separately on the local diesel generator panel) is provided in the Control Room for the cooling water system. In the unlikely event of a failure in one diesel cooling system, the backup diesel generator is available for immediate emergency use. The single failure criteria analysis of the cooling water system is shown in Table 9.5-12.

The system is housed in a seismic Category I structure. Failure of nonseismic components in the area does not prevent the operation of this system.

9.5.5.4 Inspection and Testing Requirements

Testing and inspecting is performed as described in Subsection 9.5.4.4.

9.5.6 DIESEL GENERATOR STARTING SYSTEM

9.5.6.1 Design Bases

The Diesel Generator Starting System is an air-powered system designed to start the diesel generator set. Redundant starting air systems are provided for each diesel generator set. Starting signals are listed in Section 8.3.1.1. The starting system is designed to meet the requirements of 10 CFR Part 50, Appendix A, GDC 2, 4, and 5, NRC Regulatory Guide 1.29, IEEE-387-1977, and the Branch Technical Positions MEB 3-1, APCSB 3-1, and EICSB-17 [5], [6], [7], [12], [13], [14], [16], [18].

9.5.6.2 System Description

The diesel engines are started by injection of compressed air into the cylinders. The redundant starting systems for each diesel generator are shown on Figure 9.5-55. Each diesel generator has two 100- percent-capacity air systems. Each system includes an air compressor, air receiver, an air dryer, and two solenoid starting valves (redundant solenoids for each bank). In-line air dryers are desiccant-type having activated alumina as the drying medium. Capacity is 76.1 SCFM at 250 psig and 120ºF (including purge capacity of 6 SCFM). The air dryers are designed to maintain a dew point of at least 10ºF less than the lowest expected ambient temperature.

Each receiver is sized to contain enough air for five starts. The air pressure normally available for starting is between 220 and 250 psig. The air receiver relief valve is set at 275 psig. The diesel generator will not start in EMERGENCY mode if the air pressure drops to 150 psig. Low pressure alarms are actuated if the pressure falls to 210 psig. Two air supply lines are provided, each having redundant solenoid starting valves. One valve passes enough air to start the diesel engine, thus ensuring a sufficient supply of starting air if one valve fails to operate. Each receiver also has a valve to permit periodic blowdown of accumulated moisture and foreign material.

Instrumentation is provided to monitor the operation of the system. On low receiver pressure, redundant pressure elements actuated an alarm in the diesel generator room and a trouble alarm in the Control Room.

There are no differential pressure switches on the filters. Air flows through the lines only when a diesel start is called for. At all other times, there is no flow and, consequently, no differential pressure. Normal periodic maintenance will assure that air will flow unimpeded when needed.

The starting air system is supplied with local pressure indication for each air receiver.

9.5.6.3 Safety Evaluation

Safety classification of the system meets the intent of ANSI N18.2 and N18-2a, Safety Class 3 [15] as shown on Figure 9.5-55.

The starting system provides a reliable source of starting power for the diesel generator. The system is designed to seismic Category I requirements and the worst anticipated environmental and climatic phenomena are considered in the design as discussed in Section 3.11B. To further ensure system reliability, the system components which are located off the diesel generator set are designed to the requirements of the ASME B&PV Code, Section III, Class 3. Exceptions are

the compressors, aftercoolers, air dryers and the piping up to the class break which are non-safety related. These components are designed to manufacturer's standards and are pressure tested pneumatically.

To minimize fouling of the starting air valves, or filters with contaminants, automatic drip-traps are provided on the after coolers and manual blowdown capability is provided on the receivers to collect any liquid carryover. Inline air dryers remove excess moisture. Strainers are located downstream of the receivers to catch rust.

The lines from the air receiver to the engine are sloped such that any contaminates present in the lines will collect at a low point prior to connection to the engine.

Each starting air system is provided with two redundant starting air admission valves to preclude non-operability of a single starting air system in the event of failure of one of the admission valves.

Periodic running of the diesel engine, and blowdown of the drip-traps and low points will minimize the build-up of contaminants in the starting air system.

The single-failure analysis presented in Table 9.5-13 demonstrates the ability of the system to perform its design function while subjected to a single failure.

9.5.6.4 Inspection and Testing Requirements

Testing and inspecting are performed as discussed in Subsection 9.5.4.4.

9.5.7 DIESEL GENERATOR LUBE OIL SYSTEM

9.5.7.1 Design Bases

Each diesel generator lube oil system is designed to provide adequate engine lubrication under all operating conditions, including immediate full-load operation after starting. The diesel generator lubrication is designed to meet the requirements of 10 CFR Part 50 [5], [6], [7], NRC Regulatory Guide 1.29 [12], and Branch Technical Positions APCSB 3-1[13], MEB 3-1[14], IEEE 387-1977[16] and EICSB-17[18].

9.5.7.2 System Description

The diesel generator lube oil system is depicted by Figure 9.5-56. The diesel generator operates as a dry sump system and has one main lube oil pump that is engine driven. The main pump is a positive displacement pump and draws oil from the sump through a course strainer. The lube oil travels through the pump, through the cooler, through the filter, and then through the strainer to the engine. Full-flow filtering is available. This ensures that the oil keeps its required quality during cooling. The lube oil system is supplied with the following instrumentation and alarms on the Diesel Room Control Panel to provide operating information:

1. Lube oil tank, low level alarm

2. Low temperature, lube oil in alarm

3. High temperature, lube oil in alarm

4. Low temperature, lube oil out

5. High temperature, lube oil trip

6. High temperature, lube oil out

7. High differential pressure, lube oil filter

8. High differential pressure, lube oil strainer

9. Low pressure turbo oil, right bank

10. Low pressure turbo oil, left bank

11. Low pressure turbo oil, trip

12. Low pressure engine oil alarm

13. Low pressure engine oil trip

14. High temperature bearings - trip

15. High pressure crankcase - trip

16. An on-off temperature control for the lube oil heater

17. Local pressure and temperature, indication

Alarms will be set based on the engine manufacturers allowable operating conditions. When an alarm sounds on the Diesel Room Control Panel a “DIESEL TROUBLE” signal will also sound in the control room. Personnel will be dispatched to the Diesel Generator Room to ascertain the specific problem and take required action. Alarm response procedures will be available to assist the operator in determining what action should be taken.

The lubrication oil system is cooled by jacket water. The complete system is designed and supplied by the Diesel Generator Manufacturer. The design of the lubrication oil cooler is solely based on parameters established by the Diesel Generator Manufacturer. For design parameters of the lube oil cooler, see Table 9.5-17. The jacket water system is cooled by service water. The following design information applies to the design of the jacket water heat exchanger:

1. Maximum nominal inlet temperature of 117°F,

2. Maximum allowable pressure drop of six (6) psi,

3. water velocity in heat exchanger of 8.5 fps Maximum

4. design pressure of 150 psig.

The Lube Oil System is a closed system and during normal operation, there is no way for deliterious materials to enter the system. Refilling of the Lube Oil System is accomplished through a normally capped two inch connection. Administrative controls are established as required to govern the refilling operation.

The required oil quality as specified by the engine manufacturer is maintained by automatically filtering and straining the oil as it is circulated in the engine. Administrative measures taken to maintain the required quality of the oil will be detailed in the preventative maintenance program which deals with lubrication oil quality. The lube oil will be changed, or sampled to determine if changing is required, at regular intervals based on recommendations from the manufacturer and the lubrication supplier. This will be performed as part of the preventive maintenance program using approved procedures.

The engine crankcase is protected from overpressurization by the following design features. The system is equipped with alarms and trips initiated by high oil temperature and/or pressure, which shut down the diesel generator if operating limits as specified by the engine manufacturer are exceeded. The lubrication oil system trips are blocked by safety injection actuation. The crankcase is equipped with centrifugal blowers for crankcase ventilation during diesel generator operation. The crankcase is also equipped with blowout panels to protect the unit from damage. The lubrication oil sump, the lubrication oil filter and the lubrication oil strainer are all vented.

Periodic monitoring of the level instrumentation associated with the Lubrication Oil Sump Tank may indicate an uncontrolled loss of lube oil. Also an increase in lube oil temperature or a decrease in oil pressure, or both, may be attributed to oil leakage. Any leakage discovered during routine inspections (or, investigating an occurrence described above,) will be corrected in accordance with operating and maintenance procedures.

CPNPP does not have the requirement of a timed prelube before manually starting the diesel engines.

When the engine is not operating, a motor-driven prelube pump automatically starts and draws oil from the sump, passes it through a strainer and a filter, and then into the engine lubricating system. This recirculation of lube oil continues until the engine receives another signal. The oil is heated by an electrical immersion heater in the sump. This ensures continuous prelubrication of the engine and standby heating of the oil.

The heated oil, in this keep warm mode, is circulated throughout the engine block but does not feed the turbocharger's lubrication header. Check valves in the lube oil system insure that oil is kept near these assemblies so that lubrication can be provided by the engine-driven lube oil pump within one-half of one engine revolution on startup.

9.5.7.3 Safety Evaluation

Safety classification of the system meets the intent of ANSI Standard N18.2 and N18.2a, Safety Class 3 [15] as shown on Figure 9.5-56.

The diesel generator lube oil system is designed to seismic Category I requirements.

The ANS Safety Class 3 components in the system are designed to the requirements of the ASME B&PV Code, Section III, Class 3. Component supports are designed in accordance with

ASME B&PV Code, Section III, Class 3 but fabricated as a minimum to AISC-1970. When a component is commercially unavailable as ASME B&PV Code, Section III, Class 3 design, the component is proven of equivalent quality.

"Equivalent quality", of a component is interpreted to mean an item designed for normal commercial use upgraded to ASME B&PV Code Section III requirements through seismic design, testing, qualification and documentation.

All such components are listed in Table 17A-1.

Low oil pressure to the main header is alarmed by a pressure switch. Oil pressure is indicated by a pressure gauge locally. Another pressure switch trips the engine on low lube oil pressure (if no safety injection actuation signal is present).

The volume of lubrication oil in the sump is sufficient to ensure continued operation under emergency conditions. A redundant diesel generator unit has been provided to meet the single-failure criterion, should a failure of the diesel generator lube oil system necessitate the stoppage of a diesel generator unit.

Both the high lube oil temperature and the low lube oil pressure monitoring systems have separate alarm and trip switches, since a failure of these components could result in an engine malfunction being undetected. Automatic shutdown of diesel generators is discussed in Section 8.3.1. A failure mode and effects analysis is given in Table 9.5-14.

The diesel generator lube oil system is housed in a seismic Category I structure. Failure of non-seismic components outside this structure does not affect this system. This structure also protects the system from the worst anticipated environmental phenomena as described in Sections 2.3 and 2.4.

9.5.7.4 Inspection and Testing Requirements

Testing and inspection is described in Section 9.5.4.4.

9.5.8 DIESEL GENERATOR COMBUSTION AIR INTAKE AND EXHAUST SYSTEM

9.5.8.1 Design Bases

The Diesel Generator Combustion Air Intake and Exhaust System is designed to provide the following:

1. Adequate combustion air for each emergency diesel generator under all operating conditions

2. Adequate capability to direct the engine exhaust to the outside air under all operating conditions

The system is designed to meet the single failure criterion. In order to further ensure its continued availability, the system is designed to withstand the effects of the worst anticipated environmental phenomena and seismic activity. The system is designed in accordance with 10

CFR Part 50, Appendix A, GDC 2, 4, and 5, NRC Regulatory Guide 1.29, IEEE 387-1977, and applicable codes listed in Apendix 17A [5], [6], [7], [12], [16].

9.5.8.2 System Description

The Diesel Generator Combustion Air Intake and Exhaust System is shown on Figure 9.5-57. It consists of the Air Intake System, turbocharger, turbocharger air intercooler, and the Exhaust Gas System. Each engine has its own independent Diesel Generator Combustion Air Intake and Exhaust System.

The Diesel Generator Combustion Air Intake System consists of a filter, silencer, flexible connection, adapter for connection to turbocharger intake, air intake duct connection, and interconnecting piping. The filter is capable of being cleaned during operation, and is located inside the Diesel Generator Building. The volume of intake air is sufficient to ensure continued operation under emergency conditions. The silencer is of the in-line horizontal type. The flexible connection is provided to relieve the turbocharger of any possible weight or strain due to piping weight, expansion, or contraction.

The turbocharger consists of an air compressor and a gas turbine coupled on the same shaft.

The turbocharger air intercooler transfers heat produced by the compression of combustion air by the turbocharger to the Jacket Water Cooling System.

The Diesel Generator Combustion Exhaust Gas System consists of an outlet adapter for the turbocharger gas turbine outlet, a flexible connection to ensure expansion protection and vibration isolation, a muffler to reduce the exhaust gas sound level, an exhaust duct connection, interconnecting piping, and a relief valve.

The Diesel Generator Combustion Air Intake and Exhaust System is provided with the following instrumentation:

1. A combustion air pressure indicator on engine panel for right bank and left bank of engine intake manifold

2. A digital electronic indicator for cylinder exhaust gas temperatures on engine panel; a selector switch for selecting temperature readout is also mounted on this panel.

The diesel generator combustion air intake is provided with intake air filters to protect the engine from dirt or other contaminants. The missile-protected intake, as shown on Figures 9.5-58 and 9.5-59, is located 36 ft above ground level to minimize the intake of ground dust or debris. The missile-protected inlet protects the Diesel Generator Combustion Air Intake System from external missiles and prevents the entrance of rain or snow. The combustion air intake and the exhaust are located apart so that dilution or contamination of the intake air by exhaust products does not affect operation of the diesel generators. The engine exhaust is directed upward at a point approximately 35 ft higher than the intake and 55 ft horizontal from the intake, which is directed downward. The intake is located at a sufficient distance (approximately 750 ft from the closest diesel generator) from the onsite bulk gas storage to preclude any contamination of the intake air due to an accidental onsite gas release.

The exhaust pipe downstream of the exhaust silencer contains a drain which prevents water from collecting. A full-flow relief valve located upstream from the exhaust silencer permits the engine to continue operating if the portion of the Diesel Generator Combustion Exhaust Gas System exposed on the roof is crushed by a tornado generated missile. If the normal flow path for the exhaust is blocked, this valve will discharge the exhaust gases directly to the atmosphere. There is no piping downstream of the valve.

9.5.8.3 Safety Evaluation

The Diesel Generator Combustion Air Intake and Exhaust System components, excluding interconnecting piping off the engine, are designed in accordance with manufacturers standards to perform in accordance with IEEE Standard 387 [16]. The interconnecting piping is constructed to Safety Class 3 requirements in accordance with ANSI N18.2 [15] with the exception of the flexible connectors which are Safety Class 3 but non-ASME and that piping on the roof of the Diesel Generator Building and the relief valve which are classified as NNS. The Diesel Generator Combustion Air Intake and Exhaust System components are designed to seismic Category I or II requirements as applicable, with the exception of the relief valve and the flexible connectors located on the foof. [See Table 17A-1]. See Section 3.7 for details. The Diesel Generator Combustion Air Intake and Exhaust System components are not located near any high- or moderate-energy piping systems other than these serving the same respective diesel generator. This system meets the requirements of NRC Branch Technical Positions APCSB 3-1 and MEB 3-1 [14]. The air intakes for the Diesel Generator are protected from tornado generated missiles by a labyrinth shielding arrangement as shown on Figure 9.5-59. Air enters the building from below, thus making a direct missile entrance impossible.

The exhaust piping, to a point downstream of the relief valve, is protected by a shielding structure on the roof, see Figure 9.5-58. The piping downstream of the relief valve exists this protective structure and connects to the exhaust silencer.

A redundant diesel generator unit is provided to meet the single failure criterion if a failure of the Diesel Generator Combustion Air Intake and Exhaust System necessitates the stoppage of a diesel generator unit. The exhaust silencer is located outside the seismic Category I structure and therefore not protected from external missiles. If piping and components located outside are damaged to such an extent that exhaust gases are blocked, a relief valve is provided which provides an alternate discharge path for the exhaust gases. An opening in this protective structure allows exhaust gases from the open relief valve to exit. This opening is located in such a way as to prevent a direct missile from striking the exhaust. Thus, exhaust silencer damage that blocks the exhaust causes the relief valve to open so that the system continues to operate. A failure mode and effects analysis is given in Table 9.5-15.

In addition, the two 6 inch vent pipes from the diesel engine lube oil sump tank and crankcase to the atmosphere are also partially located outside the seismic Category 1 structure. However, the two lines are protected from external missiles by a missile protection enclosure and are seismically analyzed to maintain their functional and structural integrity.

A potential fire in a diesel generator room, together with a failure of the fire protection system is considered not to significantly degrade the quality of the combustion air for the redundant diesel generator due to the following rationale:

1. Taking into account the type of potential fire hazard present, the volume of the diesel generator room and the location of the room ventilation inlet and exhaust louvers, the majority of the smoke and hot gases will rise and flow out through the room ventilation exhaust louvers.

2. The postulated fire must burn unattended for a substantial period of time before causing pressurization of the diesel generator room and subsequently forcing smoke and hot gases out through the room ventilation inlet louvers located at elevation 813'-6" (see Figures 9.5-58 and 9.5-59 for the location and proximity of these louvers).

3. The normal air intake for each diesel generator is approximately 30 percent greater than the air intake required for combustion at 100 percent load. This 30 percent higher flow capacity means that a large quantity of smoke and extraneous gas would be required to cause significant dilution of the redundant diesel combustion air before effecting the diesel generator performance.

Each diesel generator area is protected by a fixed deluge water spray system. The use of this type of system eliminates the concerns of degrading the performance of possible loss of a diesel generator due to pollution of the combustion air by the fire extinguishing medium.

9.5.8.4 Inspection and Testing Requirements

Testing and inspection are described in Subsection 9.5.4.4.

REFERENCES:

1. National Fire Protection Association Handbook, 14th Edition 1/76.

2. Appendix “A” to Branch Technical Position APCSB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants, docketed prior to July 1, 1976, dated August 23, 1976.

3. NRC Regulatory Guide 8.5, Immediate Evacuation Signal, February 16, 1973, U.S. Nuclear Regulatory Commission.

4. ANSI N2.3-1967, Immediate Evacuation Signal For Use In Industrial Installations Where Radiation Exposure May Occur.

5. 10 CFR Part 50, Appendix A, General Design Criterion 2, Design Bases for Protection Against Natural Phenomena.

6. 10 CFR Part 50, Appendix A, General Design Criterion 4, Environmental and Missile Design Bases.

7. 10 CFR Part 50, Appendix A, General Design Criterion 5, Sharing of Structures, Systems, and Components.

8. 10 CFR Part 50, Appendix A, General Design Criterion 44, Cooling Water System.

9. 10 CFR Part 50, Appendix A, General Design Criterion 45, Inspection of Cooling Water System.

10. 10 CFR Part 50, Appendix A, General Design Criterion 46, Testing of Cooling Water System.

11. Deleted

12. NRC Regulatory Guide 1.29, Seismic Design Classification Revision 2, 1976, U. S. Nuclear Regulatory Commission.

13. Branch Technical Position APCSB 3-1, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, attached to Standard Review Plan 3.6.1.

14. Branch Technical Position MEB 3-1, Postulated Break and Leakage Locations in Fluid System Piping Outside Containment, attached to Standard Review Plan 3.6.2.

15. ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, 1973. [Including ANSI N18.2a-1975 Revision and addenda]

16. IEEE 387-1977, Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations.

17. Illuminating Engineering Society (IES) Lighting Handbook, Application Volume, 1981.

18. Branch Technical Position EICSB-17, Diesel Generator Protective Trip Circuit Bypasses, attached to Standard Review Plan Appendix 7-A.

19. Comanche Peak Nuclear Power Plant Unit 1 “Fire Protection Report”

20. NFPA 10-1981, “Portable Fire Extinguishers”

21. NFPA 12A-1980, “Halon 1301 Fire Extinguishing Systems”

22. NFPA 13-1978 and 1985, “Automatic Sprinkler Systems”

23. NFPA 14-1978, “Standpipe and Hose Systems”

24. NFPA 15-1979, “Water Spray Fixed Systems”

25. NFPA 20-1983, “Centrifugal Fire Pumps”

26. NFPA 24-1973, “Outside Protection”

27. NFPA 49-1975, “Hazardous Chemical Data”

28. NFPA 72D-1975, “Proprietary Protection and Signaling System”

29. NFPA 72E-1978, “Fire Detection Systems”

30. NFPA 80-1977, “Fire Doors and Windows”

31. NFPA 6-1974, “Industrial Fire Loss Prevention”

32. NFPA 26-1976, “Supervision of Valves”

33. NFPA 30-1984, “Flammable Combustible Liquid Code”

34. NFPA 69-1973, “Explosion Prevention Systems”

35. IEEE 383-1974, “IEEE Standard for Type Test of IE Electric Cables, Field Splices, and Connections”

36. IEEE 634-1978, “IEEE Standard Cable Penetration Fire Stop Qualification Test”

37. UL Subject 992, “Test Method for Measuring the Flame Propagating Characteristics of Floor and Floor Covering Materials”

38. ASTM E 84-1976, “Surface Burning Characteristics of Building Materials”

39. ASTM E119-1976, “Fire Test of Building Construction and Materials”

40. NEL-PIA guideline titled “Basic F.P. for Nuclear Power Plants” (1976)

41. Underwriters’ Laboratory (U.L.) “Building Materials Directory” effective edition at time of design-procurement

42. NUREG-0700, “Guidelines for Control Room Design Reviews, September, 1981”.

43. ANSI B31.1, Power Piping.

44. ASTM D975-1981, “Standard Specification for Diesel Fuel Oils”

45. ASTM D4057-1981, “Standard Practice for Manual Sampling of Petroleum and Petroleum Products”

46. ASTM D1552-1979, “Standard Test Method for Sulfur in Petroleum Products (High-Temperature Method)”

47. ASTM D2662-1982, “Standard Test Method for Sulfur in Petroleum Products (X-Ray Spectrographic Method)”

48. ASTM D2276-1978, American National Standard Test Method for Particulate Contaminant in Aviation Turbine Fuels

49. ANSI N195-1976, “Fuel Oil Systems for Standby Diesel Generators”

50. Comanche Peak Nuclear Power Plant 10CFR72.212 Evaluation Report

TABLE 9.5-1
IS NOT USED

TABLE 9.5-2
IS NOT USED

TABLE 9.5-3
IS NOT USED

TABLE 9.5-4
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TABLE 9.5-10
IS NOT USED

TABLE 9.5-11
FAILURE MODE AND EFFECTS ANALYSIS OF DIESEL GENERATOR FUEL OIL STORAGE AND TRANSFER SYSTEM

Item Described	Function	Failure Mode	Cause of Failure	Effects on Subsystem	Method of Failure Detection	Effects on System
Diesel 1EG1 oil storage tank	Stores 7-day supply of fuel (approx. 15%)	Leaks	Crack, corrosion	Loss of part of 7-day supply	Level indicator, observation	None: 1EG2 available
Transfer pump 1a	Pump fuel to day tank	No output	a. Motor fails	a. None; use pump 1B	a. Level alarm	a. None; redundancy
			b. Pump fails	b. None; use pump 1B	b. Level alarm	b. None; redundancy
			c. Loss of power	c. Cannot pump oil	c. Level alarm	c. None; 1EG2 available
Transfer pump 1b	Same as pump 1a	Same as pump 1a	Same as pump 1a	Same as pump 1a	Same as pump 1a	None: redundancy none: 1EG2 available
Transfer line	Pipe fuel to day tank	Rupture	Crack, corrosion	Only 3 hr oil available [a]	Level alarm	None: 1EG2 available
Day tank	Store 3-hr fuel supply diesel	Rupture	Crack, corrosion	Loss of fuel supply	Level alarm	None 1EG2 available

a) Temporary hose or truck fill can be used, if break is upstream of check valve.

Note:

Diesel generator, Unit 1, No. 1 = 1EG1
Diesel generator, Unit 1, No. 2 = 1EG2
Diesel generator, Unit 2, No. 1 = 2EG1
Diesel generator, Unit 2, No. 2 = 2EG2

Similar failure modes on 1EG2, 2EG1, and 2EG2

TABLE 9.5-12
FAILURE MODE AND EFFECTS ANALYSIS OF DIESEL GENERATOR COOLING WATER SYSTEM

Item Described	Function	Failure Mode	Cause of Failure	Effects on Subsystem	Method of Failure Detection	Effect on System
Engine-driven jacket water pump	Pumps jacket water	No output	Mechanical ailure	Cannot pumpwater	High water temperature	None; 1EG2 is available
Jacket water heat exchangers	Heat sink for diesel service and jacket water systems	a. Leaks	Cracks, corrosion	Temperature rises	High water temperature	None; 1EG2 is available.
		b. No flow	Blockage	Temperature rises	High water temperature	None; 1EG2 is available
Jacket water piping and valves	Transports water	a. No flow	Blockage	Cannot transfer water	High water temperature	None; 1EG2 is available.
		b. Leakage	Cracks and corrosion	Temperature rises	High Water temperature	None; 1EG2 is available
Jacket water keep-warm pump	Circulates water for standby readiness	a. No output	Motor fails, pump fails, loss of power	Cannot transport water	Low water temperature	None; 1EG1 is available.
Standby electric heater	Heats water for standby readiness	No heat	Loss of power	Fails to heat water	Low water temperature	None; 1EG1 is available.

Note:

Diesel Generator - Unit 1, No. 1 = 1EG1
Diesel Generator - Unit 1, No. 2 = 1EG2
Diesel Generator - Unit 2, No. 1 = 2EG1
Diesel Generator - Unit 2, No. 2 = 2EG2

Similar failure modes for 1EG2, 2EG1, and 2EG2

TABLE 9.5-13
FAILURE MODE AND EFFECTS ANALYSIS OF DIESEL GENERATOR STARTING SYSTEM

Item Described	Function	Failure Mode	Cause of Failure	Effects on Subsystem	Method of Failure Detection	Effects on System
Air lines	Transports air	No flow	Blockage, cracks, corrosion	Subsystem not available	Low tank pressure	None: redundant start air subsystem is available for 1EG1.
Manifold	Delivers air	No flow	Blockage	Subsystem not available	As above pressure	None: redundant start air subsystem is available for 1EG1.
Compressor, Aftercooler, and Air Dryer	Refills air storage tanks	No air being supplied	Loss of power.	No refilling of air tanks	Low tank pressure	None: redundant compressor sub-system is available. For 1EG1.
Compressor	Refills starting air storage tank	Loss of air	Seismic or other pressure boundary failure	None (Safety Class 3 isolation provided)	Low tank pressure	None: (Safety Class 3 isolation provided)
Tank	Stores air for five attempted starts	Loss of air	Leaks, ruptures	No starting air	Low tank pressure	None: second tank is available. to provide air to 1EG1.
Starting Air Pressure Boundary Solenoid Operated Valve (1-SV-3421-1E)	Air receiver tank isolation	Failure to isolate the receiver	Mechanical or electrical malfunction	No starting air for DG start	Low receiver pressure	None: Redundant Start Air Subsystem is available for 1EG1.

Note:

Diesel Generator, Unit 1 - No. 1 - 1EG1
Diesel Generator, Unit 1 - No. 2 - 1EG2
Diesel Generator, Unit 2 - No. 1 - 2EG1
Diesel Generator, Unit 2 - No. 2 - 2EG2

Similiar failure modes for 1EG2, 2EG1, and 2EG2

TABLE 9.5-14
FAILURE MODE AND EFFECTS ANALYSIS OF DIESEL GENERATOR LUBE OIL SYSTEM

Item Described	Function	Failure Mode	Cause of Failure	Method of Failure Detection	Effects on System
Engine-driven lube oil pump	Pumps lube oil	Pump failure	Mechanical Failure	Low oil pressure	None: 1EG2 is available
Lube oil piping and valves lube oil	Transports	1. No flow	1. Blockage	1. Low oil level	None: 1EG2 is available.
		2. Leakage	2. Cracks or corrosion	2. Low oil pressure	
Lube oil heat exchanger	Cools oil	1. Leaks oil or water	1. Cracks or corrosion	1. Low oil level	None: 1EG2 is available.
		2. No heat transfer	2. Blockage	2. High oil temperature	
Lube oil heater	warms oil	Fails to heat	Loss of power	Low oil temperature	None: 1EG1 is available.
Pre-lube pump	Circulates oil for standby readiness	Pump failure	Motor fails; pump fails; loss of power	Low oil temperature	None: 1EG1 is available.
Strainers and filters	Cleans oil	Clogged element	Particulate matter in oil	High pressure drop	None: 1EG2 is available.

NOTE:

Diesel Generator Unit 1, No. 1 = 1EG1
Diesel Generator Unit 1, No. 2 = 1EG2
Diesel Generator Unit 2, No. 1 = 2EG1
Diesel Generator Unit 2, No. 2 = 2EG2

There are similiar failure modes for 1EG2, 2EG1, and 2EG2.

TABLE 9.5-15
FAILURE MODE AND EFFECTS ANALYSIS OF DIESEL GENERATOR (COMBUSTION AIR INTAKE AND EXHAUST SYSTEM)

(Sheet 1 of 2)

Item Described	Function	Failure Mode	Cause of Failure	Effects on Subsystem	Method of Failure Detection	Effects on System
Intake air pipes and flexible connectors	Pipe air to the engine	a. Rupture	Crack, corrosion	Loss of outdoor intake air	Visual	None: 1EG1 is still available
		b. Low flow	Blockage	Loss of adequate air	High exhaust gas temperature	None: 1EG2 available
Intake air filters	Cleans intake air	Low flow	Blockage	Loss of adequate clean intake air	High exhaust gas temperature	None: 1EG2 available
Air intake silencer	Reduce intake air sound level	a. Rupture	Crack, corrosion	Loss of outdoor intake air; excessive noise	Excessive noise	None: 1EG1 is still available
		b. Low flow	Blockage	Loss of adequate intake air	High exhaust gas temperature	None: 1EG2 is still available
Turbocharger	Provide combustion air	No air being supplied	Loss of compressor or turbine	No flow	Stopping of engine, high exhaust gas temperature	None: 1EG2 available
Turbocharger air intercooler	Cools intake air	a. Reduced heat	Fouling	Loss of adequately cooled air	Loss of engine output	None: 1EG2 available
		b. Tube rupture	Cracks or corrosion	Induction of water	Engine stops	None: 1EG2 available

TABLE 9.5-15
FAILURE MODE AND EFFECTS ANALYSIS OF DIESEL GENERATOR (COMBUSTION AIR INTAKE AND EXHAUST SYSTEM)

(Sheet 2 of 2)

Item Described	Function	Failure Mode	Cause of Failure	Effects on Subsystem	Method of Failure Detection	Effects on System
Exhaust gas pipes and flexible connection	Pipe exhaust gas to the outside of plant	Rupture	Crack, corrosion	Exhaust gas inside the Engine Room	Excessive noise	None: 1EG1 is still available
Exhaust silencer and outside piping	Reduce exhaust air sound level	a. Rupture	Crack, corrosion	Excessive noise	Excessive noise	None: 1EG1 is still available
		b. Low flow	Blockage	Exhaust relief valve opens to provide alternate flow path	Excessive Noise	None: Relief valve opens

NOTE:

Diesel Generator - Unit 1, No. 1 = 1EG1
Diesel Generator - Unit 1, No. 2 = 1EG2
Diesel Generator - Unit 2, No. 1 = 2EG1
Diesel Generator - Unit 2, No. 2 = 2EG2

Similar Failure Modes For 1EG2, 2EG1, and 2EG2

TABLE 9.5-16
THIS TABLE HAS BEEN DELETED

TABLE 9.5-17
LUBE OIL COOLER DESIGN PARAMETERS

Design heat removal rate, Btu/hr	3.38×10^6
Required heat removal rate, Btu/hr	3.22×10^6
Flow rate (tube side), gpm	900
lb/hr	438,300
Flow rate (shell side), gpm	500
lb/hr	225,153

	Shell	Tube	Diff.
Inlet temperature	185.0	147.2	37.8
Outlet temperature	156.4	154.7	1.7
Temperature differential	28.6 (drop)	7.5 (rise)	

Design margin in heat removal rate:

The design heat removal rate of 3.38×10^6 Btu/hr includes an excess of 0.16×10^6 Btu/hr over the required heat removal rate.

TABLE 9.5-18
JACKET WATER COOLER DESIGN PARAMETERS

Design heat removal rate, Btu/hr	25.5×10^6
Required heat removal rate, Btu/hr	17.0×10^6
Flow rate (shell side), gpm	1202.6
lbs/hr	608,000
Flow rate (tube side), gpm	1439.84
lbs/hr	0.7205×10^6

	Shell	Tube	Diff.
Inlet temperature F	175.0	115.0	60
Outlet temperature	147.0	138.6	8.4
Temperature Differential, F	28.0 (drop)	23.6 (rise)	

Design margin in heat removal rate:

The design heat removal rate of 25.5×10^6 Btu/hr includes an excess of 8.5×10^6 Btu/hr over the required heat removal rate.

10.0 STEAM AND POWER CONVERSION SYSTEM

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10.1 SUMMARY DESCRIPTION

The steam and power conversion system for each of the two CPNPP units includes one steam turbine generator and all of the auxiliary equipment and support systems necessary for its operation. Each turbine generator unit is independent of the other, as there are no shared components. The following description applies to each of the units.

The turbine is a tandem-compound, four-flow, 1800-rpm machine installed on the Turbine Building operating deck (elevation 830 ft 0 in.) and is supported by the turbine pedestal.

Steam is supplied to the two-flow, high-pressure turbine from four steam generators, which in turn are supplied with heat from the primary loops of the Pressurized Water Reactor (PWR).]

The high-pressure turbine exhaust is directed through two moisture separator/reheaters (MSR) operating in parallel. The steam is dried and reheated in the MSRs and then passed through the two two-flow low-pressure turbines.

Exhaust steam is condensed in a two-shell, single-pressure, single-pass, surface-type condenser. Condensate storage capacity in the condenser hot wells is equivalent to approximately five minutes operation at the maximum steam flow rating. The Condensate and Feedwater System returns feedwater to the steam generators through six stages of extraction feedwater heating.

Heat balances reflecting the original design at an NSSS rated thermal power of 3628 MWt (maximum guaranteed rating) are shown on Figures 10.1-1 and 10.1-2, respectively. The rating of the electric generator is 1,410,000 kVA, 60 Hz, 0.90 power factor.

The description contained within this chapter is a general description of the original design of the steam and power conversion system. The actual operational parameters (fluid flow rates, temperatures, pressures, etc.) may differ from those reflected throughout Chapter 10 due to as-built conditions, steam generator blowdown, and physical and operational changes made to improve net heat rate and maximize electrical power generation at the licensed reactor thermal power.

Uprates in the reactor licensed thermal power of either unit may also be implemented (See Section 1.1.4). An evaluation of the available margin within the steam and power conversion systems is performed as part of any formal unit uprate to determine the extent that the steam and power conversion system's original design margin can support operation at such power levels. Any modifications to those systems are reflected herein consistent with the level of design detail contained in Chapter 10.

The following system diagrams are shown in the FSAR Flow Diagram Volumes:

Flow Diagram	
M1-0202	Main Steam, Reheat & Steam Dump System
M1-0204	Main Condenser Piping Diagram

M1-0211	Main Condenser Evacuation System (Condenser Vacuum)
M1-0223	Turbine Gland Sealing System
M1-0210	Circulating Water System
M1-0240	Circulating Water Chlorination System
M1-0244	Condensate Cleanup System (Condensate Polishing)
M1-0204	Condensate System
M1-0203	Steam Generator Feedwater System
M1-0239	Steam Generator Blowdown Cleanup System
M1-0206	Auxiliary Feedwater System
M1-0205	Extraction Steam System
M1-0207 thru 0209	Heater Drain System
M1-0212	Turbine Plant Cooling Water System
M1-0213	Auxiliary Steam System
M1-0214	Turbine Oil Purification System
M1-0243	Plant Gas Supply (Nitrogen and Hydrogen)
M1-0222	Secondary Plant Sampling System
M1-0228	Process Sampling System
M1-0228	Post Accident Sampling System

The following system diagrams are shown in Section 10.4:

Figure	
10.4-1	Main Condenser Piping Diagram – Plan
10.4-2	Main Condenser Piping Diagram - Elevation
10.4-12	Flow Diagram - Auxiliary Feedwater Failure Mode Analysis
10.4-22	Main Feedwater Loop Seal Arrangement
10.4-23	Unit 1 Integrally Grooved Tube Sheet Pressurization System

Circulating water for the condensers is provided from the Squaw Creek Reservoir (SCR), where heat is primarily rejected to the atmosphere by surface evaporation. Makeup water for the reservoir is furnished from Lake Granbury.

The safety-related features of the steam and power conversion system include portions of the main steam supply system, upstream from and including the first moment restraint beyond the main steam stop valves used for Containment isolation. Included in these portions are the main steam safety valves, atmospheric steam dump power relief valves, main steam line condensate removal drain pots up to and including the pneumatically operated isolation valves, sections of the Steam Generator Blowdown and Process Sampling System up to and including pneumatically operated isolation valves, which form part of the Containment isolation scheme, and the air vents on the safety class portion of the main steam lines. The steam lines supplying the auxiliary feedwater pump turbine driven are also safety-related. The safety-related portions of the Steam Generator Feedwater System include those sections downstream of and including check valves just upstream of the feedwater isolation valves and those portions of the chemical feed, nitrogen supply, and process sampling systems which form part of the Containment pressure retaining boundary. The Auxiliary Feedwater System is also safety-related.

Instrumentation, controls, and alarms are provided to monitor system functions, as described in detail in subsequent sections.

TABLE 10.1-1
MAJOR STEAM AND POWER CONVERSION EQUIPMENT SUMMARY

Component	Maximum Guaranteed Rating Unit 1	Unit 2
Steam Generator[a]		
Steam flow rate (total 4 steam generators), lb/hr	16.26×10^6	16.26×10^6
Feedwater inlet temperature, °F	450.3	450.3
Steam outlet temperature, °F	545.1	543.2
Steam outlet pressure, psia	1005	989
Steam generator reactor coolant inlet temperature, °F	620.4	620.4
Steam generator reactor coolant outlet temperature, °F	558	558
Turbine Generator[b]		
Steam to turbine generator, lb/hr	14,774,975	14,715,715
Throttle inlet pressure, psia	980	948.94
Throttle enthalpy, Btu/lb	1192.3	1192.1
Percent moisture	.21	.4
Power output, kW	1,263,444	1,253,440
Power factor	0.90	0.90
Hydrogen pressure, psig	75	75

a) Reference: WCAP-16840

b) Reference: Unit 1 - WB-10909-1 (Figure 10.1-1)
Unit 2 - WB-10939-1 (Figure 10.1-2)

10.2 TURBINE GENERATOR

10.2.1 DESIGN BASES

The turbine generator unit is a tandem-compound, four-flow, single reheat-type, direct-connected unit, supplied by Allis-Chalmers Power Systems Inc./Siemens Power Corporation. The turbine generator is intended to be base loaded. The turbine receives steam from four steam generators (thermal energy) and converts the thermal energy to mechanical energy through rotation of the turbine shaft. The turbine, in turn, is directly connected to an electric generator, which produces electrical energy upon rotation against an excited field. The turbine generator original design basis is to produce 1,160,706 kWe, when operating, with 15,140,016 lb/hr of saturated steam at 975 psia, 541.5 F, 0.38-percent moisture at the throttle, 3.5 in. Hg absolute at the exhaust. The turbine consists of one high-pressure element and two low-pressure elements. The maximum original guaranteed design basis heat balance condition is as shown on Figure 10.1-1 of Section 10.1.

The Unit 1 turbine generator is designed to produce 1263.4 MWe when operating at 14,774,975 lb/hr saturated steam at 980 psia and 1192.3 Btu/lb with 0.21% moisture at the throttle and 1.414 in Hg absolute at the turbine exhaust. The Unit 2 turbine generator is designed to produce 1253.4 MWe when operating at 14,715,715 lb/hr saturated steam at 948.9 psia and 1192.1 Btu/lb with 0.40% moisture at the throttle and 1.414 in Hg absolute at the turbine exhaust.

The turbine generator includes an electrohydraulic control system which is capable of adjustment to step load changes as required.

The steam turbine has the following loading capabilities:

1. The capability to accept step changes in load of 10 percent rated power and ramp changes of 5 percent rated power per min in the range from 40- to 100-percent full load

2. The capability to accept 50-percent load reduction from rated power without reactor trip. This capability is based on the capability of secondary plant systems to accept 40-percent steam dump and to maintain adequate feedwater supply to the steam generators. The reactor accepts 10 percent of the load rejection during this mode of operation.

3. The capability to accept complete load rejection from the maximum rated power level with reactor and turbine trip

4. The capability to follow generator-demanded load changes automatically when these load changes occur within the automatic control range of 15- to 100-percent full power

The turbine generator and all its appurtenances generally meet the applicable requirements of the following codes, standards, and legislation:

1. American Insurance Association (A.I.A.)

2. American National Standards Institute (ANSI)

3. American Society of Mechanical Engineers (ASME)

4. American Society for Testing and Materials (ASTM)

5. American Welding Society (AWS)

6. Heat Exchange Institute (HEI)

7. Hydraulic Institute (HI)

8. Institute of Electrical and Electronics Engineers (IEEE)

9. National Electrical Safety Code (NESC) ANSI C2

10. National Electrical Manufacturers Association (NEMA)

11. National Fire Protection Association (NFPA)

12. Occupational Safety and Health Act (OSHA)

13. Steel Structures Painting Council (SSPC)

14. Tubular Exchanger Manufacturers Association (TEMA)

Main steam and feedwater piping outside the Containment from the end of the moment restraint in the Safeguards Building to the Turbine Building is designed in accordance with ANSI B31.1. Main steam piping between the moment restraint in the Safeguards Building and the Containment wall and the piping located within the Containment is designed in accordance with Section III of the ASME B&PV Code. Feedwater piping between the moment restraint in the Safeguards Building and the Containment wall and the piping located within the Containment is designed in accordance with Section III of the ASME B&PV Code. Seismic classification conforms to Section 3.2, and the seismic design requirements are defined in Section 3.7.

The turbine-generator unit is not designed for operation under the stresses that could be imposed by the half safe shutdown earthquake (1/2 SSE) or the safe shutdown earthquake (SSE). However, the turbine generator is designed to function under the thermal stresses that could be imposed because of upset conditions, emergency conditions, and faulted conditions as defined in Section 2 of ANSI N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plant, August 1973.

10.2.2 DESCRIPTION

10.2.2.1 Turbine

The turbine is a multicasing, tandem compound, four-flow, reaction type, 1800-rpm unit with 46-in. last-stage blades. No radiation shielding is required.

The turbine consists of one double-flow, high-pressure element in tandem with two double-flow, low-pressure elements. Two horizontal cylindrical external combined moisture separator/reheaters (MSRs) are included; one is installed on the operating deck on each side of the turbine at elevation 830 ft 0 in. Exhaust steam from the high-pressure turbine element enters the MSR shells where moisture is removed, and high-pressure main steam passing through

tubes in the reheater heats the lower-pressure steam prior to its entering the low-pressure turbines. Each MSR normally processes half of the steam flow.

The MSRs are pressure vessels containing heat exchanger tube bundles and moisture-separating elements. The single stage reheater consists of U-tube type tube bundle with a fixed tube sheet of carbon steel clad with Inconel and integrally finned tubes of ferritic stainless steel. The moisture separators are of the Chevron type, and are constructed of stainless steel for durability in the wet steam environment.

The MSRs are designed for high reliability with external tube header, welded tube-to-tube sheet joints, ample drainage capability, and provisions for supports with thermal flexibility. Safety relief valves and heating steam supply valves are also supplied.

Steam from the steam generators is supplied to the high-pressure turbine through four pipes, each containing a stop valve in series with a control valve combined into a single compact casing. The valve units are located above the operating deck (elevation 830 ft 0 in.), close to the high-pressure turbine, with two units on either side of the turbine. This location minimizes the entrapped steam volume on valve closure, thus limiting potential overspeed following a load rejection. The above-floor arrangement and turbine rollaway cover simplify valve inspection and maintenance.

The stop valve is opened by control fluid pressure and closes rapidly by spring force when tripped. The control valve is positioned by a hydraulic servomotor but also closes rapidly by spring force after loss of control fluid pressure.

Upstream from each stop and control valve is a permanently installed steam strainer body which includes removable steam strainer.

Steam from the MSRs enters the low-pressure turbine through low-pressure stop and control valves located at each low-pressure turbine inlet. Each of these valves has its own hydraulic actuator. The low-pressure stop valves are operated by the trip system. Low-pressure control valves are operated by the control system. These valves limit overspeed and maintain unit stability on loss of load.

Provisions have been included to allow for automatic testing of the high-pressure and low-pressure stop and control valves while the turbine is in operation.

The turbine generator is equipped with all instrumentation and control devices required to operate the unit. Included is an electrohydraulic-type speed and load control system designed to control speed and load of the unit under the load changes indicated in Subsection 10.2.1. The system has a main electrohydraulic governor to control speed during normal operation with load reject logic to limit top speed of the unit to a value below the emergency overspeed trip setting. With the exception of the hardware overspeed system sharing three speed probes with the governor system, the overspeed systems are completely independent of the governor control system and initiate a trip at an overspeed of approximately 110 percent to immediately close all main steam stop and control valves and the low-pressure turbine stop and control valves. A trip of the turbine generator will initiate a signal which will activate the reactor trip system when greater than 50% reactor power (P9 interlock).

10.2.2.2 Lubrication System

Bearings are lubricated during operation by oil from the main shaft-driven oil pump located in the front pedestal of the turbine. Three half-capacity oil coolers are provided, two active and the other standby, with mechanically interlocked four-way transfer valves on the inlet and outlet to ensure safe onload changeover from one cooler to the other. Dual filters with a changeover valve are provided in the oil pipe which supplies the combined journal-thrust bearing and the shaft-oil lifting pumps.

The bearing oil-supply piping is guarded against fire hazard resulting from leaking by enclosure in steel ducts. Access is provided for inspection of the piping. The oil drain piping is separate from the supply piping.

Three half-capacity auxiliary oil pumps, an emergency bearing oil pump, and a shaft-oil lifting pump are provided for startup, shutdown, and turning-gear operation.

10.2.2.3 Turning Gear

The turbine generator is equipped with an oil-hydraulic turning gear consisting of an impulse turbine wheel solidly connected to the turbine shaft and segmental nozzle boxes mounted within the front bearing pedestal. Oil from the auxiliary oil pump header is supplied to the turning gear through two motor-operated valves and is discharged through the nozzles, thus impinging on the wheel and causing the turbine-generator shaft to turn. In the case of startup from a stationary condition, the initial friction of the shaft in the journal bearings is overcome by the shaft-oil lifting pump which is fed from a high-pressure oil pump into the bottom of the journal bearings.

The oil-hydraulic turning gear turns the turbine generator at a speed in excess of 80 rpm. The relatively high speed ensures that good, stable lubrication films exist in the journal bearings while they are on turning gear. Also, the speed helps the shaft and moving blades cool down evenly after shutdown and circulates the air or steam in the turbine casings to minimize any temperature differences between upper and lower stationary parts, thus preventing thermal distortion of either rotors or casings. Because there are no gear teeth to be engaged or disengaged, the turning gear is not susceptible to damage. Furthermore, the turning gear requires neither mechanical nor electrical interlocking with the shaft-oil lifting system, which eliminates all requirements that the unit coast down to a stationary condition before engaging the turning gear.

10.2.2.4 Steam Sealing System

The steam glands which seal the high-pressure turbine casing-to-rotor clearances are of the axial-flow labyrinth type. Each gland consists of a large number of radial sealing strips of 10 mil thickness at the seal tip that are caulked into grooves in the stationary gland rings. Each stationary ring is composed of six segments. These are spring-mounted in horizontally split gland ring holders. The shaft glands of the low-pressure turbines use similar sealing strips caulked into spring-mounted segmental gland rings; however, the low-pressure shaft is smooth where it passes through the glands to accommodate the relatively large axial expansions of the low-pressure rotors. Each shaft seal has two annular chambers. One is connected to the sealing steam header, and the other is exhausting steam mixed, with a small amount of air leakage to a seal steam condenser.

In the normal operating range, the sealing steam header is supplied from the high-pressure turbine gland leakoffs and crossaround system.

For startup and at low loads, an auxiliary source such as main steam is required. The seal steam header pressure is regulated at slightly above atmospheric pressure by a supply valve and a leak valve, both operated by an pneumatically controlled system that includes a pressure transducer. The leak valve discharges to the lowest pressure feedwater heater. A smaller amount of air and sealing steam are continuously drawn off the outermost section of each gland and discharged to the seal steam condenser, which is maintained at slightly below atmospheric pressure by a motor-operated exhauster. The control system includes provisions for manual operation from the Control Room, with an automatic balancing device providing for switchover from automatic to manual without a sealing pressure transient.

The glands are designed to avoid local overheating of the shaft if abnormal conditions result in a rub at any seal.

In the event of rubbing of either the moving or the stationary sealing strips, the spring-mounted segments yield against the pressure of the springs associated with each segment. If this radial movement is not sufficient to relieve the rub, the thin strips are worn down with only a slight amount of heat, thus avoiding shaft damage from excessive local overheating. If necessary, glands can be easily repaired. This is done by removing the worn strips and caulking new ones into the same grooves.

In the case of the high-pressure turbine, access is gained to the outer gland rings by removing the horizontally split gland ring holders from the ends of the turbine outer casing. If the glands have been damaged, the sealing strips in the ring segments are renewed without having to open the high-pressure turbine casing. The shaft glands of the low-pressure turbines are enclosed within horizontally split gland ring holders attached rigidly to the adjacent bearing pedestals; consequently, they can be inspected or repaired without disturbing the low-pressure turbine outer casings.

Caulked-in radial sealing strips (similar to those used in the shaft glands) also minimize steam leakage over the shrouds of moving and stationary blade rows. Generally, three strips are used to seal each row of shrouded blades. The sealing effect is improved in the moving rows by using stepped shrouds with two different sealing-strip diameters. Each continuous shroud has an integral elevated or depressed cylindrical surface, and strips of different heights (corresponding to the shroud steps) are caulked into circumferential grooves opposite the shroud.

10.2.2.5 Steam Extraction Connections

Turbine steam extraction connections are provided for six stages of feedwater heating and for the MSRs as discussed in Section 10.4.10.

10.2.2.6 Generator

The generator is rated at approximately 1,410,000 kVA, is three-phase, and has a 60 Hz frequency at 0.90 power factor. The generator stator windings and bushings and generator rotor windings are water cooled, and the core and other components are hydrogen cooled at a maximum pressure of 75 psig. The generator is integrated and connected with its own excitation

system. Generator rating, temperature rise, and insulation class are in accordance with ANSI C50 and NEMA MG 1, as applicable.

10.2.2.6.1 Hydrogen Storage and Supply

Hydrogen for cooling the generator is stored outdoors, about 300 feet west of the Turbine Building. (See Figure 1.2-1). The piping arrangement of this facility is shown on Figure 10.4-19, Flow Diagram Plant Gas Supply System-Hydrogen Supply, and described in Section 10.4.15.

Gas supplied from the storage tanks passes through a pressure-regulating station located at the tanks and is distributed through the Turbine Building at approximately 150 psig. Secondary pressure regulators control the gas supply to the individual generators at a maximum pressure of 65 psig. The remote location of the outdoor bulk storage tanks minimizes the hazards of fire or explosive concentrations of hydrogen. In the event of a break in the supply line, the faulted line is automatically isolated by the excess flow control valve.

In the event of a gross supply line rupture, the flow rate across the excess flow valve would go critical, causing an immediate closure. Reaction times are a function of the hydrogen flow rate from the line rupture - typical reaction times are less than one (1) second.

The excess flow control valve in the hydrogen supply system operates on the following principle. Hydrogen flow through the excess flow valve creates a pressure drop proportional to the square of the flow rate. A spring holds a seat plug off the seat. When the pressure drop across this seat plug reaches the set point of the valve, the force against the upstream side of the seat plug overcomes the spring tension and forces the seat plug against the seat, forming a bubble tight seal. Differential pressure holds the valve closed until a manual bypass valve is opened and the pressure equalized across the valve.

10.2.2.6.2 Fire Prevention and Protection

1. Fire Prevention

   Following manufacturing, the empty generator stator frame with attached shields and terminal box is subjected to a hydraulic test to ensure that it will be capable of withstanding maximum hydrogen explosion pressures.

   The use of hydrogen as a coolant in the generator requires special safety equipment to ensure that hazardous operating conditions will not occur. As a precaution against the explosion hazard the air inside the generator is neither directly replaced with hydrogen during generator filling nor the hydrogen replaced directly with air during the emptying procedure. In both cases the generator is scavenged with an inert gas, argon.

   A meter system serves for measuring the H_2 content of the cooling gas in the generator as well as the composition of gas mixture (argon/air and H_2/argon) during filling and emptying of the generator. A second mechanical purity meter system is also supplied.

   Particular precautions are taken with respect to a H_2 leak in the generator bearing compartment. A special vapor exhauster creates a slight vacuum in the bearing compartment. Any hydrogen collecting in the bearing compartment is drawn off by the

exhauster and vented to atmosphere. If the exhauster fails, a second exhauster is automatically started.

To prevent the hydrogen which enters the bearing compartment from escaping via the oil drain pipes, the drain oil is admitted into the turbine oil tank via the Hydrogen Degassing Storage Tank and a loop seal. This loop seal is permanently filled with oil to prevent the escape of gas.

The seal oil drained from the seal oil tank passes into the Hydrogen Degassing Storage Tank to which the vapor exhauster is also connected. The exhauster creates a slight vacuum in the Hydrogen Degassing Storage Tank so that oil saturated with hydrogen is degassed.

2. Fire Protection

For discussion on fire protection, see Section 9.5.1.

10.2.2.7 Automatic Controls

As described in Subsection 10.2.2.1, steam from the steam generators is supplied to the high-pressure turbine through four pipes, each containing a stop valve in series with a control valve combined into a single compact casing. The stop valve is opened by control fluid pressure, and, when tripped, closes rapidly by spring force. The control valve is positioned by a hydraulic servomotor and also closes rapidly by spring force after loss of control fluid pressure.

Steam from the MSR units enters the low-pressure turbines through low-pressure stop and control valves at each low-pressure turbine inlet. Each of the valves has its own hydraulic actuator which is operated by the trip and control system, respectively. The valves operate as low-pressure stop and control valves and help limit overspeed and maintain unit stability on loss of load.

The standard turbine control system is a redundant electrohydraulic control (EHC) system with load rejection logic to mitigate an overspeed event. After a full load rejection the unit will experience a speed transient up to a maximum of 110 percent of rated speed, then it will settle down to a steady-state speed corresponding to 60.3 Hz at idling (no-load) operation. The digital EHC system provides a frequency influence (droop) function which can be switched on or off from the OM computer work stations.

10.2.2.7.1 Speed and Load Control With EHC

The EHC contains a speed control loop and a load control loop. The speed control is used primarily to bring the turbine up to synchronous speed. However, the speed control always remains in operation over the entire load range. The load control is switched on automatically when a generator breaker is closed.

The speed control compares the actual turbine speed with a reference setting and uses the difference as a control signal. The actual speed is derived from an electrical speed measuring system that contains speed probes mounted in the front bearing pedestal and directed to a toothed wheel. The toothed wheel is mounted on the turbine shaft within the front high-pressure bearing housing. As the toothed wheel turns, a signal of varying frequency is induced in the

probes proportional to the turbine speed. The digital signals are converted to analog signals in the EHC cabinet. There are three speed signal channels in this system. If any channel fails, it is automatically switched out, and an alarm is given; however, operation continues with the remaining channels. The load control compares the actual generator load with a reference load setting and delivers the difference as a load-demand error signal. The actual generator load is obtained from three load transducer channels using generator voltage and current inputs. The range, deadband, and droop of the frequency influence function of the digital EHC system is adjustable from the OM computer workstations.

During load pickup, the automatic transfer from speed to load control at synchronization/generator output breaker closure is smooth because a balance control is incorporated into the system.

If load is rejected to less than approximately 15 percent, the speed/load control signal is overridden, initiaing fast closure of all control valves. Without adjustment of the reference speed setting, the system can control a full-load rejection to no-load with a residual speed deviation of about 0.75 percent. It can control a load rejection to auxiliary load with less than 0.4 percent residual speed deviation.

The EHC system contains internal self-monitoring equipment for critical functions.

The value of the EHC speed/load signal positions the turbine control valves through two redundant valve lift controllers, amplifiers, proportional valves and followup piston banks. Adjustments are provided to ensure the proper control of high pressure and low pressure valve operation.

10.2.2.7.2 Hydraulic Control Fluid System

The hydraulic portion of the control system uses fire-resistant pure phosphate ester fluid at a maximum pressure of 455 psig. This moderate pressure level helps minimize the possibility of leaks, and the fluid reduces the possibility of a fire even if a leak should occur.

The system includes a control fluid tank with three half-capacity pumps, two active and one standby. The standby pump is automatically activated if the fluid pressure falls below a preset level. Two full-sized fluid coolers are provided, one active and one standby, with mechanically interlocked three-way valves for safe onload transfer to the standby cooler, if necessary.

The fluid is maintained clean by continuously circulating a portion of the tank volume through filters arranged in series. A separate small motor-driven pump provides this circulation. All fluid returning to the tank passes through sieve filters located at the return connection inside the tank. Dual filters are also provided at the control signal inlet to each control valve actuator.

10.2.2.7.3 Overspeed Protection

The important function of overspeed protection is provided by both the redundant EHC control system and the redundant overspeed protection systems.

The EHC system serves as the primary device for mitigation of an overspeed event. Load rejection logic built into the system monitors the generator output. If the system senses a

negative gradient over a certain load range, the protection logic will override the control logic and quickly close both the HP and LP control valves for a preset time.

If for some reason the EHC control system was unable to prevent an overspeed event, two independent and diverse electronic overspeed protection systems will trip the turbine at 110 percent of rated speed (1980 rpm). The overspeed trip systems consist of two independent and diverse trip circuits called hardware (HW) and software (SW) overspeed trip systems. Either system will independently trip the turbine and each is configured for 2 of 3 trip logic.

The hardware overspeed trip system consists of a primary hardware trip circuit and a backup software trip circuit sharing three speed probes. The three speed probes are connected to three individual speed measurement cards that compare the speed signal to preset setpoints. If the setpoints are exceeded (<10 rpm and >1980 rpm), a trip signal is sent from each speed measurement card to the hardware and software trip circuits.

The HW trip signals are then processed by three circuit cards with board mounted relays. The three cards work together as one unit to check for a 2-of-3 trip coincidence. The three circuit cards bypass the fail-safe turbine trip system and de-energize the three relays on the output of each of the three channels of the turbine trip system, and consequently de-energize the turbine trip block. As a backup, the HW trip signals for the speed measurement cards are also processed by three channels of the fail-safe turbine trip system, and de-energize the output to the turbine trip block.

Independent of the hardware overspeed system a separate software overspeed trip system is provided serving as an independent and diverse overspeed trip. The software overspeed trip is independent and utilizes separate speed probes, speed measurement cards, fail-safe software-based trip system and is installed in a separate cabinet with independent power supplies.

As with the hardware system, three speed probes feed three individual speed measurement cards that compare the speed signal to preset setpoints. If the setpoints are exceeded (<10 rpm and >1980 rpm), a trip signal is sent from each speed measurement card to the software trip circuits. The trip signals are then processed by the three channels of fail-safe software-based turbine trip systems. Each channel is configured with 2-of-3 trip logic. The outputs of the trip systems will de-energize which will trip the turbine trip block.

Rapid closure of all turbine stop and control valves will result from either the HW or SW overspeed trips. Only one of the system is necessary to trip the turbine. Since the HW and SW overspeed trip systems are independent and actuation of either system will cause a turbine trip, only one of the two systems is necessary to trip the turbine. Since the HW and SW overspeed trip systems are independent and actuation of either system will cause a turbine trip, only one of the two electrical overspeed trip systems is required to be operable.

Proper operation of the HW and SW overspeed trip systems are checked online daily with an automatic test initiated by the protection system. The turbine trip block test is performed every 14 days. The HW overspeed trip relays and their initiating relays cannot be tested at power. These relays will be tested each outage to ensure operability.

10.2.2.7.4 Turbine Stress Evaluator

The turbine stress evaluator continuously monitors differential temperatures in critical areas of the turbine. It automatically computes and provides operating limits for safe speed and load changes.

The turbine stress evaluator uses special dual thermocouples which measure the temperature at the wall center and at the inner wall surface of the first main steam stop valve. The temperature of the high-pressure turbine admission inner wall surface is measured to evaluate the high-pressure rotor thermal stresses. From these measured temperatures it calculates actual thermal stress conditions and compares them with allowable conditions considering the prevailing speed or load. A selection circuit determines which of the critical areas has the smaller margin between actual and permissible temperature differentials. The magnitude of the limiting margins, which are converted into temperature and load allowances, is displayed on Operating and Monitoring (OM) screens at computer workstations in the Control Room.

The limiting values calculated by the turbine stress evaluator and displayed on its indicator are used as input signals to the speed load reference limiters of the EHC equipment, thus closing the loop between the turbine stress evaluator and the turbine control system. Consequently, speed or load changes are only possible within the allowable ranges, as determined and indicated by the turbine stress evaluator.

Temperature values, speed, load, and temperature and load allowances are displayed on the OM screens. If a malfunction occurs, it is still possible to operate the turbine generator with the turbine stress evaluator completely out of service. This is done by using data on the OM screens and criteria provided in the operating instructions.

10.2.2.7.5 Turbine Stop and Control Valves

A trip signal from any designated source causes closure of the stop valves in each of the four high-pressure steam admission lines, the control valve in each high-pressure steam admission line, the low-pressure turbine valves (of which there are two per low-pressure steam admission line), the motorized stop and power-assisted check valve combination in steam extraction lines 1, 2, 3, and 4. Steam extraction line 3 branches off from the cold reheat line. Extraction lines 5 and 6, located at the low-pressure end, are not valved, since they are inaccessible inside the condenser neck and contribute little to overspeed potential because they are low-pressure lines.

The high-pressure stop valves are simple two-position valves which are held open against spring force by control fluid pressure. Loss of this pressure causes rapid closure.

The control valves are also opened by fluid pressure and closed by spring force. They close rapidly on trip command in a manner similar to the stop valves. The stop valves are designed to be leaktight when closed.

Both the stop and control valves have stems, sealing rings, and other sliding surfaces which are designed and manufactured for reliable operation. The stem seals are of the labyrinth type with radial clearance between the stem and sealing rings. This radial clearance is provided as insurance against the possibility of valve sticking caused by galling or buildup of material deposits on sliding surfaces.

The low-pressure stop and control valves in each low-pressure turbine steam admission are of butterfly-type design. Each valve has its own actuator. The low-pressure stop valves are two position valves which close rapidly on receipt of a trip command. The low-pressure control valves are equipped with actuators similar to the high-pressure control valves for position control in response to a control signal.

The high pressure (HP) stop and control valves have closing times of less than 500 millisecond and the butterfly-type low pressure (LP) stop and control valves less than 1500 millisecond.

Four HP control valves are arranged one in each of four HP admission pipe to the Turbine. An independent two position (open-close) HP stop valve is located immediately upstream of the HP control valve. The four HP stop valves and four HP control valves are in series forming a one-out-of-two system in each of the four admission lines. No interconnection between the four separate HP stop valves and control valves is installed for throttle controlled operation with full-arc admission of the HP turbine.

Each admission pipe to an LP turbine is equipped with a butterfly-type stop valve and a butterfly-type control valve in series forming one-out-of-two system in each of the admission pipe. No interconnection between these admission valves is installed.

Overspeed is limited by rapid closure of the turbine control or stop valves whenever turbine power exceeds generator output, as would exist immediately after a load reduction. The normal control system is designed to limit transient overspeed to less than 110% of rated speed after a full load rejection by closing both the HP and LP control valves. The control system consists of a redundant electrohydraulic control (EHC) system and built-in load rejection logic, as described in section 10.2.2.7.1.

In the unlikely event that the EHC control system should fail, the HW and SW overspeed systems (described in 10.2.2.7.3) will act to limit the overspeed to less than 120% of rated speed. The two independent and diverse electronic trips meet the intent of SRP 10.2 Part III, for redundance and independence. An analysis of the electronic overspeed systems is included in an attachment to the “Probability of Turbine Missiles” provided in reference 2. Either the HW or SW overspeed system will de-energize the three trip solenoids on the turbine trip block, draining turbine trip fluid and resulting in rapid closure of all turbine stop and control valves. Only one of the two systems is needed to trip the turbine, with each system utilizing 2-of-3 trip logic.

The turbine overspeed protection system thus consists of three separate and independent sub-systems, any one of which is capable of limiting the overspeed to within acceptable limits.

Therefore, the failure of any one turbine stop or control valve will have no effect on the overspeed protection system because the valves are arranged in series.

The extraction line isolation valves are located close to the turbine to minimize the volume of trapped steam.

The turbine trip signal provides the closing signal for the controlled extraction steam power assisted non-return check valves in the extraction steam piping to heater Nos. 1, 2, 3, and 4. This protects the unit against excessive overspeed due to reverse flow of steam from the extraction system through the turbine. The turbine trip signal will vent the air from the actuator

cylinder of these non-return check valves and the spring loaded piston will supply the energy to close the valve within one (1) second to prevent overspeed of the turbine.

Heaters 5 and 6 are located in the condenser neck with the channel end extended out of the condenser. Non-return check valves for these heaters are not provided because: 1) Anti-flash baffles are provided to restrict the reverse flow from these heaters to a sufficiently low flow so that it cannot adversely affect turbine overspeed, and 2) These lines do not contain enough available stored energy to overspeed the turbine.

10.2.2.7.6 Testing and Trip Devices

During normal operation, the performance of the turbine trip device can be checked by an automatic turbine tester. The automatic turbine tester is a software-based system built into the turbine protection system and is initiated through the operating control screen.

The automatic turbine tester (ATT) is capable of testing the following components:

1. Turbine Trip Block (TTB):

The operator starts the turbine trip block test from the control screen. Tests are completed in the shortest possible time by the use of automatic sequential testing. Safety of operation is ensured because only one trip block piston is tested at a time. A trip will not occur because the turbine trip block requires two pistons in the tripped position to drain turbine trip fluid. The test sequences until each trip piston is tested and then exits automatically. Any failures are annunciated on the operator annunciator screen. These tests are normally conducted at least once every two weeks using the ATT.

2. High-pressure and low-pressure stop and control valves

If the unit load is less than 986 MWe the high-pressure stop and control valves can be tested. Stop valve closure times are measured by the automatic turbine tester, and any excessive time causes a warning to be displayed to the operator. In this way either proper operation is verified or potential valve sticking is detected during periodic valve testing. Additional monitoring sensors have been installed on the HP turbine stop and control valves. The data from these sensors can be trended to detect valve closing time degradation as input to scheduled maintenance. These tests are normally conducted at least once every 26 weeks using the ATT or manual test.

10.2.2.8 Other Equipment and Features

In addition to the electronic overspeed trips, the turbine is equipped with a manual trip pushbutton at the front of the turbine, a remote pushbutton, two low-vacuum trip systems, an MSR high level trip device, a trip for thrust bearing failure, and a low lube oil pressure trip device. Additional protective features include atmospheric relief diaphragms in the low-pressure turbine casings, a water spray system to avoid excessive exhaust temperature, and alarms for high lube oil temperature, low condenser vacuum, and low control fluid pressure.

A turbine control system supervises the turbine speed and control valve position, bearing and shaft vibration, shaft and casing expansions, shaft eccentricity, and turbine and bearing metal temperatures.

Operating and Monitoring (OM) computer workstations are supplied in the Control Room. These workstations provide indication and alarm as well as computer screen-based controls of the EHC, Seal Steam Control (SSC), Turbine Stress Evaluator (TSE), Generator Temperature Control (GTC), Leakage Water Return Control (LRC), and (MSR) Heating Steam Control Systems, and an automatic turbine tester (ATT).

Thermal insulating material is furnished, and sound-absorbing metal appearance lagging is supplied for enclosing the high-pressure turbine, including the main stop and control valves.

10.2.2.9 Turbine Protective Devices and Alarms

Turbine protective devices and Control Room annunciations are provided as follows:

1. Thrust bearing failure detectors for trip and alarm

2. Atmospheric relief diaphragm mounted in each low-pressure turbine outer casing

3. Water spray system to avoid excessive exhaust temperature

4. Extraction line check valves to protect the turbine from overspeed as a result of reverse flow after a load rejection turbine trip; these check valves also minimize the possibility of water induction because of high water levels in the heaters.

5. Alarm for high lubricating oil temperature

6. Low lubricating oil pressure trip and alarm

7. Alarm for high low-pressure turbine casing temperature

8. High vibration alarm

9. Low-pressure turbine exhaust low vacuum trip and alarm

10. MSR high water level trip and alarm

11. TSE failure

12. EHC failure

13. SSC failure

14. High Moisture in Generator Gas

10.2.2.10 Turbine Trips

Turbine protective trips cause closing of all the turbine admission valves and initiate closure signals to the swing-type turbine extraction check valves. After the closing of turbine stop valves or release of control fluid pressure, a signal is generated for initiating the reactor trip (if greater than 50% reactor power), the steam dump control, and the electrical generator trip. The reactor trip signals are initiated by instrumentation which provides input to the reactor protection system.

Turbine trip is initiated by any of the following:

1. Reactor trip
2. Steam generator high-high level or safety injection
3. Overspeed (two channels)
4. Generator trip
5. Condenser low vacuum (one mechanical and two redundant electrical)
6. Manual turbine generator trip (from Control Room)
7. Manual turbine generator trip (at turbine)
8. Thrust bearing failure
9. Low bearing oil pressure
10. Moisture separator high level (each MSR)
11. Loss of hydraulic control fluid pressure

Electrical generator trip is initiated by the following:

1. All of the turbine trip signals, including loss of lube oil pressure and thrust-bearing failure
2. Generator protection signals, including signals from the main and unit auxiliary transformers and the 345 kV Switchyard breakers.
3. Manual generator trip from Control Room
4. Electronic generator protection (EGP), including signals from primary water flows, primary water temperature, and primary water tank level.
5. Rotor ground protection
6. Pilot exciter short

After an automatic turbine trip, the generator breaker trip is delayed to furnish uninterrupted power to the reactor coolant pump motors for at least 30 sec without relying on the success of a bus transfer, provided the generator conditions permit this. Likewise, following a manual turbine trip, or a turbine trip due to certain turbine faults or certain generator protection signals, the generator trip is delayed approximately 11.5 seconds. Generator trip (after a turbine trip) is also conditional upon detection of reverse power, except for certain generator faults, to minimize the probability and the degree of overspeed after a turbine trip. The trip logic is shown functionally on Figure 7.2-1, Sheet 16 and Figure 10.2-1.

For a narrative and schematics, which describe in detail the sequence of events in a turbine trip, see Reference 3.

For response times the following is provided. These times assume normal control mode of operation which is electrohydraulic control (EHC).

Approx. Time (Milliseconds)	Action
0	Load rejection
10	Load rejection is sensed by the electrical control system.
20	Output signal of the valve lift controller starts to decrease.
40	Piston of the electr. Hydr. Converter which equals actual lift value starts to move into (valve) closed position.
50	Secondary fluid pressure for HP control valves starts to decrease
80	Secondary fluid pressure for LP control valves starts to decrease.
150	HP control valves start to close.
250	LP Control valves start to close.
300	HP control valves closed.
1250	LP control valves closed.

10.2.2.11 Turbine Supervisory Instrumentation

Turbine supervisory instrumentation is provided for the monitoring of the following conditions:

1. Shaft eccentricity
2. Bearing and shaft vibration
3. Shaft and casing expansions
4. Control valve position
5. Turbine speed
6. Turbine metal temperature
7. Bearing metal temperature
8. Generator hydrogen gas and stator cooling water temperature
9. Exhaust hood temperature

10. Condenser vacuum

11. Stator winding temperature

12. Turbine stress evaluator indicator

13. Temperature load allowance turbine stress evaluator

10.2.2.12 Safety Considerations

As described in Section 7.2.1.1.2 the turbine generator hydraulic pressure switches and the stop valve limit switches are part of the anticipatory reactor-trip-on-turbine-trip channels which provide turbine tripped signals to the Reactor Protections System.

Failure in a high or moderate energy piping or failure of the connection from the low pressure turbine to condenser could damage components of the anticipatory reactor trip channels because no protection is provided for the above mentioned switches. The safety analysis does not take any credit for the anticipatory reactor trip as described in Section 15.2. Consequently, there are no safety implications if the switches are damaged as a result of a pipe break.

10.2.3 TURBINE DISK INTEGRITY

This section provides information demonstrating the integrity of the turbine disks and rotor, as failure of these components can result in turbine high-energy missiles as discussed in Section 3.5.1.3. As described in Reference [10], for the purposes of missile formation the most hazardous missiles are generated from the low pressure turbine. The potential missile energy from the high pressure turbine is less than that from the low pressure turbine because of its much smaller potential missile mass and thicker turbine casing. Therefore, the high pressure turbine potential missiles are bounded by the low pressure turbine potential missiles.

10.2.3.1 Materials Selection

Each two-flow, low-pressure turbine rotor is made from a stepped shaft with a total of eight shrunk-on blade disks arranged in symmetrical groups of four. The material for disks 1, 2, 3, and 4 has the German standard designation of 26 NiCrMoV 145, which is 3.5-percent nickel-alloy steel similar to ASTM A471.

The percent nominal chemical compositions are as follows:

Item	26 NiCrMov 145
Nickel (Ni)	3.5
Chromium (Cr)	1.5
Carbon (C)	0.26
Manganese (Mn)	<0.40
Vanadium (V)	0.15 (max.)

Item	26 NiCrMov 145
Molybdenum (Mo)	0.4

The mechanical properties (maximum value for tensile strength, minimum for all other values) at 68°F (20°C) are as follows:

Item	Disc 1	Disc 2	Disc 3	Disc 4
Tensile Strength, Ksi (Mpa) 0.2 percent offset	<147 (<1010)	<141 (<970)	<154 (<1060)	<154 (<1060)
Yield Strength, Ksi (Mpa)	113-123 (780-850)	109-119 (750-820)	119-129 (820-890)	119-129 (820-890)
Elongation (L/d=5), Percent	>15	>15	>15	>15
Reduction of Area, Percent	>50	>50	>50	>50
Impact Strength, ft-lb (J) (Average of three Charpy V-Notch Specimens)	>100 (>130)	>100 (>130)	>100 (>130)	>100 (>130)
FATT, °F (°C)	<-110 (<-80)	<-110 (<-80)	<-110 (<-80)	<-110 (<-80)

The compressive residual stress level of the heat-treated disk forgins, as measured by the ring core process, shall not be lower than 14,504 psi (100 Mpa) and higher than 36,260 psi (250 Mpa) in the middle of the hub area; shall not be lower than 29,007 psi (200 Mpa) and higher than 58,015 psi (400 Mpa) in the area of the rim of the hub; shall not be lower than 14,504 psi (100 Mpa) and higher than 36,260 psi (250 Mpa) in the rim area. Residual tensile stresses are not permitted.

The maximum disk stress at the shrink fit is required to remain less than 50 - 60% of the yield strength of the material at nominal conditions. Disk failure in this speed range can only occur if serious material defects exist or if a major design or manufacturing error is made.

The disk forgings are produced from vacuum-degassed alloy steel and are heat treated for an optimum combination of high fracture-toughness throughout the disk volume and for calibrated compressive residual stresses at the hub bore surface.

10.2.3.2 Fracture Toughness

The safety analysis of each disk design is based on the principles of linear elastic fracture mechanics (LEFM). The methodology used is described in Reference 10.

10.2.3.3 High-Temperature Properties

As described in Subsection 10.2.3, the high-pressure rotor cannot generate missiles; consequently, no discussion of the stress-rupture properties need be presented.

10.2.3.4 Turbine Disk Design

The turbine generator is designed and tested for safe operation up to 120 percent of rated speed, including operating speed and all maximum overspeed excursions that can occur during normal operation of the unit. In this speed range, rotor failure can only occur as a result of material defect or a major error in the design or manufacture of the rotor.

10.2.3.5 Preservice Inspection

Each low-pressure disk is subjected to comprehensive design, manufacture, and quality assurance to ensure its reliability throughout the life of the turbine-generator unit as discussed in Reference [2].

The low-pressure disk forgings are produced from vacuum-degassed alloy steel and are heat treated for an optimum combination of high fracture-toughness (throughout the disk volume) and calibrated compressive residual stresses at the hub bore surface. Each disk forging is examined by ultrasonic testing as follows:

1. In the rough as-forged condition before heat treatment

2. Premachined with contours prior to heat treatment for mechanical properties

3. After heat treatment for mechanical properties

Ultrasonic testing equipment and techniques are in accordance with DIN Standards 54 120 and 54 122 and can detect and measure flaws as small as 0.04 in. (1 mm) in equivalent diameter.

The evaluation of ultrasonic inspection is based upon reported indications as follows:

1. Isolated single indications of an equivalent defect size of 0.2 in. (5 mm), in accordance with The Distant Gain Size diagram, and larger

2. All isolated indications causing a decrease of more than 10 percent of the back reflection

3. All indications of the linear type of the area type, as well as clustered indications, regardless of the size of the single indications in the cluster area

4. All indications of defects located within a 2-in. (50 mm) zone surrounding the axial center bore

Material samples are taken from each low-pressure turbine disk forging near the hub bore surface for the determination of the NDTT as a fracture-toughness criterion and of the 0.2-percent offset yield strength as a strength criterion. The results of these mechanical tests, in

combination with the ultrasonic testing and residual stress measurement results, are decisive for the acceptance of the forging.

Before machining out the hub, the following test results of the fully heat-treated forging are obtained and scrutinized:

1. Results of ultrasonic test, covering 100 percent of the disk volume, including documentation of all indications required

2. Actual chemical composition of the forging material at specified locations

3. Tensile and drop-weight (NDTT) test results at a specified location

After machining to the dimensions of the order drawing, the bore of the forging is magnetic-particle-tested. In addition, residual stresses are measured by the ring core method described in Reference [2].

By careful control of heat treatment, desirable residual stress characteristics can be built into the disks. These characteristics are verified by measurements at six specified points, before and after machining, for the first of a batch of similar disk forgings. Subsequent disks of the same batch are checked at only one point.

The low-pressure turbine rotor disk and couplings are shrunk onto the shaft with the shaft in the horizontal position and with rotation of the rotor during the process. This procedure provides the following benefits:

1. Defined heat transfer from disk to rotor, even during the initial stage of the shrinking process when the clearance is still greater than zero, because the disk always rests with its weight on the shaft

2. Exact positioning of the disk by the axial compressing device during the initial stage of the shrinking process, when the clearance is still greater than zero and turning gear is in operation

3. Easy correction of rotor runout by stopping the turning gear for a calculable period of time during the initial stage of the shrinking process with the shaft in the correct position

4. Axial and radial runout checks during the entire shrinking process with the shaft in operating position

Before delivery, each completed, bladed, low-pressure rotor is balanced and subjected to an overspeed test at 125 percent of rated speed for 2 min, at a minimum temperature of 59°F. All other turbine and generator rotors are also subjected to a 120-percent overspeed test.

10.2.3.6 Inservice Inspection

The inservice inspection program for the turbine generator consists of the following:

At least one high-pressure stop and control valve is dismantled to allow visual and surface examinations of the valve shaft, disc and stem. A visual inspection of the accessible portions of

the valve shaft and disc for at least one low-pressure stop valve and one low-pressure control valve are inspected from within the associated piping. If an inspection uncovers unacceptable flaws such as cracks or excessive corrosion, further investigation of valves of that type shall be made as necessary to assure additional inspections commensurate with inspection requirements of the higher safety class valves of ASME Section XI, Subsection IWB, Paragraph IWB-2430. Inspection of one valve of each type is performed at least once per 40 months. All valves are inspected within the 10 year interval of inservice inspection.

The electronic overspeed systems are continually tested on line automatically or can be initiated manually.

As described in Subsection 10.2.2.7.8, the high-pressure stop and control valves and the low-pressure turbine valves are exercised at least every 26 weeks using the automatic turbine tester or manual test.

The turbine includes an inservice inspection program to provide assurance that disk flaws (which could lead to brittle failures at design speeds) are detected. This inspection includes disassembly of the turbine at approximately 12-yr intervals during plant shutdowns coincident with the inservice inspection schedule required by Section XI of the ASME B&PV Code. The program includes visual and surface examinations and the latest volumetric examinations as required. All normally inaccessible parts are inspected, including couplings, coupling bolts, shafts, low-pressure blading, low-pressure disks, and high-pressure rotors.

To preclude the possibility of a failure of a low pressure turbine due to cracking in the turbine disk, an ultrasonic inspection of the disk bore and keyway areas [4] is performed for each low pressure turbine at intervals not to exceed 100,000 operating hours [10].

Inplace visual examination of the turbine assembly at accessible locations is conducted during refueling shutdowns at intervals not exceeding three years.

10.2.4 EVALUATION

The turbine generator and its related steam-handling system is designed in accordance with design criteria based on latest experience with large light water reactor turbines and erected in a manner to eliminate leakage at all joints.

It is possible that activity in the secondary side of the steam generator will occur. This activity is a function of the tube leakage from the primary side of the steam generator.

The reactor coolant circulates from the reactor core, where it removes heat from the fuel elements, to the steam generators, and back to the core. In the steam generators, heat from the pressurized reactor coolant is transferred across metal tube walls to the secondary coolant to generate steam. The steam passes through the turbine(s), is condensed, and returns to the steam generators.

A small percentage of the fuel elements can exhibit small pinholes or cracks over the lifetime of the station, allowing the diffusion of radioactive fission products from the fuel into the reactor coolant. This is discussed in Section 11.1. The reactor coolant is continuously purified in the Chemical and Volume Control System (CVCS). The filters and demineralizers remove a large portion of the iodine, other fission products, and radioactive corrosion products.

A steam generator tube leak coincident with leakage from failed fuel rod is the only condition that could result in fission products entering the secondary coolant system. If a leak exists, some radioactive reactor coolant is transferred to the secondary system. In addition to the steam generator leak rate, the secondary coolant system activity is affected by moisture carryover, blowdown rates, and partition factors in the steam generator and condenser.

The activity in the reactor coolant is a function of factors such as fuel defect level, system volumes, purification flow rates, and CVCS removal factors.

Once a determination of these variables is made, the secondary side activity is calculable as a function of this leak rate. The SGBS (described in Section 10.4.8) and water chemistry (described in Section 10.3.5) limit this activity by acting in conjunction to control the chemical composition of the steam generator secondary-side water. This control is accomplished by a combination of varied blowdown rates and chemical feed, which result in reduced admission of iodine gas into the turbine generator.

The hypothetical turbine overspeed failure is described in reference 2 and 3 Section 2. The reference 3 report expands on the function of the turbine generator speed control system based on diagrams and drawings. The report also defines the response of the various speed control systems during various possible failure modes. Valve closing and test intervals are defined.

The results of a failure mode and the effect analysis for each overspeed protection system is provided in reference 2.

In the event of a high or moderate energy piping failure, it could be possible that the turbine could lose the electrical speed control system. Assuming this event were to occur with a control fluid piping failure, a failure of the electrical speed control system would have no impact on turbine speed because the loss of control fluid pressure would result in closure of the turbine stop and control valves. The turbine will be protected from overspeed by two independent and diverse channels, trip redundant, failsafe electronic overspeed systems. Any signal failure from any two speed channels or power failure to any two trip solenoids due to piping failure will also trip the turbine.

The system design includes pneumatically operated positive closing non-return check valves to provide protection against the possibilities of overspeed of the turbine. The valves are automatically closed by a turbine trip within one second from the initiation of the reverse flow. The valves are air piston operated and close by venting air.

Heaters 5 and 6 (see Figure 10.4-13) are not provided with positive closing check valves since the extraction steam lines do not contain enough available energy to contribute overspeed of the turbine.

Refer to FSAR Section 10.4.10 for a description of the function of these check valves.

The activity concentrations in the area of the turbine generator are ascertained to be low enough to classify the area as Zone I, as defined in Section 12.1.1. No shielding or controlled access is required.

Further discussion of releases to the Turbine Building and to the environment is given in Section 11.1.

REFERENCES

1. Allis-Chalmer's Power Systems, Inc., Engineering Report No. ER-503, Turbine Missile Analysis for 1800 rpm Nuclear Steam Turbine-Generators with 44-inch Last Stage Blades, July 1975.

2. Engineering Report No. ER-504, Probability of Turbine Missiles from 1800 r/min Nuclear Steam Turbine-Generators with 44-inch Last Stage Blades October 1975.

3. Allis-Chalmer's Power Systems, Inc., Engineering Report No. ER-601, Speed Control of 1800 rpm Steam Turbine - Generators for Light Water Reactor Applications, May, 1976.

4. Utility-Power Corporation Engineering Report ER-8401, "Ultrasonic Disk Inspection", August, 1984.

5. Utility-Power Corporation Engineering Report ER-8402, "Probability of Disk Cracking Due To Stress Corrosion", August, 1984.

6. Utility-Power Corporation Engineering Report ER-8102, Revision 1, "Critical Crack Sizes of Comanche Peak LP Turbine Disks", November, 1981.

7. Deleted.

8. Deleted.

9. Deleted.

10. CT-27331, Revision 5, "Missile Probability Analysis Methodology for TXU Generation Company LP, Comanche Peak Units 1 and 2 with Siemens Retrofit Turbines", Siemens Westinghouse Power Corporation, January 18, 2007.

11. TP-03143, "Missile Analysis Methodology for GE Nuclear Steam Turbine Rotors by the SWPC," July 31, 2003.

10.3 MAIN STEAM SUPPLY SYSTEM

10.3.1 DESIGN BASES

The main steam supply system is considered part of the Main Steam, Reheat, and Steam Dump System. It conveys steam from the outlet of the steam generators to the various system components throughout the Turbine Building. The steam is primarily used for driving the main turbine and for heating service in the MSRs. In addition, it is used for various auxiliary services such as the following:

1. Steam generator feed pump turbine

2. Auxiliary feed pump turbine

3. Steam dump system

4. Turbine shaft seal system

5. Plant process steam system

Steam from the outlet of the four steam generators (a total of 15,140,016 lb/hr, at 1000 psia and 0.25 percent moisture for the original plant design conditions) is routed in the main steam piping to the turbine. The piping is sized for a pressure drop of less than 25 psi. The design pressure-temperature rating of the main steam piping is 1200 psig at 650°F. The full-load nominal original design steam flow from the reheaters to the feedwater pump turbines is approximately 242,472 lb/hr at 152.1 psia. The steam supply from the main steam lines to the auxiliary feedwater pump turbine is in the range of 60,297 to 9050 lb/hr at 1236 to 100 psia steam generator pressures, respectively.

The main steam supply system is used for plant cooldown. This is achieved by progressively lowering the pressure of the steam generators; the decay heat and the sensible heat are removed by the generation of steam. When steam generator pressure has been reduced to 100 psia, the Residual Heat Removal (RHR) System is placed in operation.

The main steam supply system is designed to meet all applicable requirements of 10 CFR Part 50. This system is designed in accordance with NRC Regulatory Guides 1.26 and 1.29.

The main steam safety valves are rated to pass 105 percent of the engineered safeguard design (ESD) steam flow at a pressure not exceeding 110 percent of the main steam system design pressure (ESD flow equals 104.5 percent of the original guaranteed 3425 MWt). These safety valves prevent overpressurization of the system as specified in the ASME B&PV Code.

The steam generator atmospheric relief valves pass at least 10 percent of the steam flow used for the plant design at no-load steam pressure (1107 psia).

The main steam isolation valves (MSIVs) are designed to stop flow from either direction within five sec after receipt of signal to close to prevent uncontrolled steam release from more than one steam generator.

The steam dump system design bases are given in Section 10.4.4.

The main steam lines from the steam generator, out through the Containment, and up to the first moment restraint beyond the MSIV, are designed in accordance with the ASME B&PV Code, Section III, Code Class 2. Beyond this moment restraint, the main steam lines to the main turbine and the lines to the feedwater pump turbines, the condensers (steam dump), and the reheaters are non-nuclear-safety-related and are designed in accordance with ANSI B31.1, Power Piping.

The lines to the auxiliary feedwater pump turbine are designed in accordance with the ASME B&PV Code, Section III, Code Class 3.

The MSIVs, integral bypass valves, and bypass piping are of Code Class 1 design (The applicant has optionally upgraded this equipment to Code Class 1.) The safety valves, and atmospheric relief valves are of Code Class 2 design. The energy storing components for the MSIV operators are not within the scope of ASME B&PV Code, section III, subarticles NA-1120 & NA-1130. However, these components for the operators will comply with ASME B&PV code section VIII.

The piping is insulated, and where exposed to the outdoor environment, it is suitably protected from the weather.

The steam generator shell and lines which emanate from the steam generator shell side are barriers against release of containment atmosphere during and after a LOCA.

Steam is conveyed from the steam generators to the main turbine by four steam lines. Upstream from the MSIVs, each line is provided with five spring-loaded safety valves and one atmospheric relief valve. It is essential that the heat load be evenly shared between the four loops of the Nuclear Steam Supply System (NSSS). To achieve this, the four main steam lines are interconnected by a pressure equalizing header downstream of the steam generator isolation valves.

The sizes and routing of the main steam piping are such that under design flow, a pressure drop of approximately 22 psi is obtained between the steam generator nozzle and the turbine stop valve inlet with a steam velocity of approximately 115 ft/sec. Each steam line uses a drain system to remove accumulated condensate from the line.

To ensure steam supply to the auxiliary feedwater pump turbine (even in the steam generator isolation), two separate steam supply lines are provided from two main steam lines.

A direct connection is provided downstream of the isolation valves for the startup of the steam generator feedwater pump turbines.

Table 10.3-1 shows the design bases of the main steam piping.

The environmental design bases are given in Section 3.11.

The inservice inspection requirements are given in Section 6.6.

The inservice inspection program for the turbine generator is described in Section 10.2.3.5.

10.3.2 DESCRIPTION

The main steam supply system and interconnected piping are shown schematically on Figure 10.3-1. The nuclear safety class is applied to the main steam lines from the steam generator, up to and including the first moment restraint beyond the MSIVs located outside the Containment. The nuclear safety class is also applied to the blowdown and process sampling lines from the steam generator, up to and including the pneumatically operated isolation valves. (See Figure 10.3-1, sheet 1 of 2.) The non-nuclear-safety class portion includes the remaining portion downstream of the moment restraint. (See Figure 10.3-1, sheet 2 of 2.)

Each main steam line has a number of branch-off lines located downstream of the main steam isolation valve. All these branch-off lines are provided with steam shut-off valves as listed in Table 10.3-11.

10.3.2.1 Safety Valves

The main steam line from each steam generator is provided with five ASME B&PV code-certified, spring-loaded, double-outlet-designed safety valves (see Figure 10.3-1) to protect against overpressurization of the main steam supply system. These valves are designed to pass a total rated relieving capacity sufficient to prevent a pressure rise greater than 10 percent above system design pressure under any anticipated pressure transients.

The safety valves are installed outside the Containment, downstream of the power-operated relief valves, and upstream from the MSIVs; they discharge to the atmosphere through 18-in. dual-outlet-designed umbrella-type vent stacks.

To avoid lifting during pressure transients, set pressures for safety valves are as high as possible within the requirements of the codes. To prevent chattering during operation of the safety valves, the individual valves in each steam generator bank are set at a different pressure.

The lowest safety valve setting is 1185 psig, which is the design pressure of the main steam supply system. The highest safety valve setting is 1235 psig, which is 105 percent of the lowest safety valve set pressure minus the 10-psi piping pressure loss between the steam generator nozzle and the safety valve.

The resultant total relieving capacity of the safety valves exceeds the maximum steam generator rated flow. With a 3-percent pressure accumulation when the safety valves are operating, the maximum pressure while relieving is 103 percent of the highest safety valve set pressure (1235 x 1.03 + 15 = 1287.5 psia). This prevents the pressure from exceeding a 110-percent system design pressure (Article NC 7000 of ASME B&PV Code, Section III).

The actual capacity of any one safety valve does not exceed the flow rate specified by the steam generator manufacturer at the design pressure of the main steam supply system; this limits excessive core reactivity insertion.

Table 10.3-2 shows the design bases of the main steam safety valves.

Mounting of the exhaust annulus piping surrounding the safety valve exhaust line allows for vertical and horizontal displacement of the main steam piping caused by thermal expansion. The

arrangement of the steam discharge line allows for piping growth to prevent contact with the safety valve body.

Adequate provisions are made in the steam piping for the installation and support of the safety valves, with consideration being given to static and dynamic loads when operating and when subject to seismic shock.

10.3.2.2 Atmospheric Relief Valves

An atmospheric relief valve (ARV) is mounted adjacent to the five spring-loaded safety valves in the steam outlet piping from each steam generator. The atmospheric relief valve is of the modulating type, air operated, and fail closed, with a maximum full stroke time of 20 seconds in the automatic mode. The valve is set to open at approximately 1125 psig, 60 psi below the lowest safety valve set point. The relief valves discharge to the atmosphere and are used to avoid, wherever possible, the opening of the lowest set safety valves.

The atmospheric relief valves provide means for removal of heat from the NSSS to the atmosphere during periods when the condenser is not in service. They are sized to remove the reactor decay heat generated following a reactor shutdown. The atmospheric valves modulate open to relieve excessive pressure in the main steam lines. The pressure control set point can be adjusted with a control board hand auto station. These valves can be manually opened or closed at the valve by using the valve handwheel, or they can be remotely opened or closed by using the control board hand auto station. In the manual control mode from the main control room or hot shutdown panel, the ARVs are capable of opening/closing maximum full stroke within 40 seconds. The ARVs are required to have the capability to be operated remotely from the control room following a safe shutdown earthquake coincident with the loss of offsite power.

In Unit 1 the Westinghouse steam generators model D-4 have been replaced with a newer model. Two new solenoid valves are installed for each ARV. These solenoid valves are supplied from the opposite train power supply. These solenoid valves are operated from a key locked hand switch in the main control room to open the ARVs.

During a Unit 1 steam generator fault, if normal DC power is lost to the Train A or B buses, the ARVs can still be opened if required from the opposite train power to cool and depressurize the reactor and terminate primary to secondary leak.

Since the CPNPP Instrument Air System is not safety related, safety-related air accumulators are provided to operate the ARV's. The valves fail closed on loss of air or electric signal. Valve positions, open or closed, are indicated with control board lights.

Failure of the relief valves to open causes the system pressure to rise to the set point of the first safety valve, which would then open, preventing further pressurization of the system. The atmospheric relief valves do not provide main steam supply system overpressure protection. This overpressure protection is provided entirely by the safety valve system described previously.

The capacity of each Atmospheric Relief Valve is required to be consistent with the following requirements:

1. Sufficiently large to allow the plant to be cooled from no-load temperature to the RHR cut-in temperature of 350° F prior to the time that the condensate storage tank is exhausted.

2. Less than the maximum allowable relief capacity of a main steam safety valve.

3. Sufficiently large to allow the plant to be cooled and depressurized so as to terminate the primary-to-secondary break flow following a steam generator tube rupture event prior to the time the affected steam generator completely fills with liquid.

4. Sufficiently small such that the calculated radiological offsite dose consequences of a steam generator tube rupture are within the guidelines of 10CFR100.

The valves discharge to the atmosphere and are designed to operate over the steam pressure range of 100 to 1300 psia. Each valve inlet pipe is provided with one manual isolation valve for maintenance.

Table 10.3-3 shows the design bases of the atmospheric relief valves.

10.3.2.3 Main Steam Isolation Valves

10.3.2.3.1 General

Each main steam line is provided with a quick-acting isolation valve, and is designed to stop flow from either direction after a steam line break (five sec after receiving the closing signal) to prevent uncontrolled steam release from more than one steam generator. The valves are installed outside the Containment, downstream of the safety valves, and are provided with a manual 4-in. bypass valve for warming the system and equalizing the pressure across the isolation valve. The bypass valve is locked closed during power operation. The MSIVs can be opened manually by the operator in the Control Room without opening the bypass valve.

Each MSIV is provided with a two-train module, three-position control switch mounted on the main control board. The switch has an electrical two-train module so that valves can be closed even if one train fails. The three switch positions are close, auto, and open, with spring return to auto position. Each MSIV also has a two-train module test switch to enable a valve to be closed to a 10 percent-closed position when tested. In addition to these control board mounted switches, there is a trip switch for each of the two trains, either of which can be used to trip all four MSIVs simultaneously. Trip switch positions are close, auto, and reset, with spring return to auto.

The MSIV's are automatically closed on high-high containment pressure or steamline break protection logic (as indicated by high steam pressure rate or low steamline pressure). High steam pressure rate is only effective when steamline SI is manually blocked during startup and cooldown, and low steamline pressure is only effective when the block is removed (see Figure 7.2-1 sheet 7). The MSIV's are closed by operation of the MSIV valve actuators. The actuator is, in effect, a hydraulic cylinder coupled directly to a nitrogen-accumulator. The accumulator is designed as a chamber concentric to the hydraulic cylinder, and it stores the energy required for closing the MSIV in the form of compressed nitrogen gas. Because the accumulator is an integral part of the cylinder, the loss of any external manifolding or system elements will not prevent the actuator from closing the valve. A hydraulic control system which

maintains hydraulic fluid below the valve actuator piston is utilized to regulate valve closure velocity. Extension of the actuator to close the MSIV is accomplished by redundant 1E electric signals which operate two solenoid valves in the hydraulic control system portion of the actuator. These valves permit the hydraulic fluid below the actuator piston to flow into a hydraulic reservoir at a controlled rate as the compressed nitrogen extends the actuator to close the MSIV.

Each component of the hydraulic control system whose presence or function is required to effect the fail-safe extension of the actuator is redundant with a second component capable of performing the required function regardless of the state of operation or failure of the other. Two hydraulic control system manifolds are provided, each of which is capable of providing valve closure capability independently of the other. The MSIV fails closed on a loss of hydraulic fluid.

The initiation and control of main steam isolation is redundant and electrically and physically separated. There is no single failure in the initiation and control portions of the system that will prevent a “main steam isolation” signal from arriving at its destination.

A main steam isolation signal will close the MSIVs.

A main steam isolation signal will also isolate the flow from the drain pots upstream of the MSIVs associated with each steam line. These valves have two train inputs so that valves can be closed even if one train fails and also fail closed on either loss at electric signal or air failure. (See Section 10.3.2.7).

Following a main steam isolation signal, the main turbine will trip on either redundant reactor trip or turbine trip signals, and the closed turbine stop valves will serve as a backup to the MSIVs. Thus, in the event of a failure of a MSIV to close, main steam will still be isolated. There will be an insignificant loss of steam through the drain pots downstream of the MSIVs and an additional loss as the steam dumps to the condenser open to relieve a steam pressure spike after a turbine trip, but the steam generator will not blow down even with a MSIV that failed to close. (See Section 15.1.5)

(For a description of initiation logic, see Figure 7.2-1 Sheet 8 and 7.3.2.4.2).

Each MSIV can be tested with its own control board switch (two-train module, two-position test, or normal with position maintained). If the valve switch is in the test position and the valve does not reach 10-percent closure after a time delay, an alarm is actuated in the Control Room. When the valve is tested, it closes slowly by energizing a test solenoid in the hydraulic circuit as well as energizing the trip-close solenoid. An automatic MSIV trip-close signal overrides the test signal and closes the valve quickly.

Alarms are actuated when MSIVs have low hydraulic oil pressure or low actuator gas pressure. Each valve has position-indication lights on the main control board for open, closed, and test positions. There are monitor lights which light on valve-closed position.

The automatically operated MSIVs serve only a safety function and are not required for power operation. They are required to limit uncontrolled flow of steam from the steam generators in the event of a break in the piping system. These valves operate under the following situations:

1. Break in the Steam Line from One Steam Generator Inside the Containment Building

If the break is within the Containment, steam is discharged into the Containment. The other steam generators act to feed steam through the interconnecting header into the broken line and then into the Containment. A steam line break results in a significant pressure rise in the Containment so that reverse flow protection is necessary to prevent discharge of more than one steam generator. According to calculations, reverse flow must be interrupted to limit the Containment pressure rise to an amount below design pressure. To achieve this, the automatic isolating valves close within five sec from receipt of the initiating signal. Closure of these valves allows for a single failure of an active component.

2. Break in the Steam Line Outside Containment Building and Upstream from the Isolation Valve

   In this case, Containment Building pressurization is not a concern. However, the uncontrolled blowdown of more than one steam generator must be prevented. The 5-sec valve closure time established previously satisfies the requirements for this situation.

3. Break in the Steam Line or Header Downstream of the Isolation Valve

   The closure time established previously meets the requirements for this situation.

4. Steam Generator Tube Rupture

   In this case, a fast-acting valve closure is not required. The isolation valves limit primary coolant leakage during shutdown by isolating the damaged steam generator after the primary system pressure is reduced below the steam generator shell-side design pressure.

   Opening the MSIVs in Mode 5 following an outage and maintaining them open throughout Mode 4 and into Mode 3, to warm up the steamlines up to the turbine stop valves is acceptable. The MSIV Bypass valves may be used as described in Section 10.3.2.4, below, if the MSIVs are closed during heatup.

10.3.2.3.2 MSIV Design Requirements

1. Analysis of Accident Conditions

   The MSIVs are designed to withstand the conditions created by a large steam line break on either side of them. The valves are designed to withstand the effects of high mass flow rate, moisture carryover, and high fluid velocity. These valves have a controlled speed of closure. The surge pressure caused by the dynamic effect of valve closure does not affect the pressure boundary, in accordance with the calculation of ASME B&PV Code, Section III. The design and construction of the MSIVs ensure positive closure against flow in either direction under the following conditions:

   a. Pipe Break - High Moisture Content (Outside the Containment)

      Flow conditions are based on the relief area downstream of the MSIV that results in a maximum mass flow rate and a maximum moisture content (approaching 96 percent).

The following tabulated mass flow rate is a function of the flow restrictor throat diameters and the steam generator no-load pressure (reservoir pressure at initiation of transient):

Flow Conditions	
Steam generator outlet no-load pressure, psia	1107
Static pressure at valve minimum flow area, atmospheric pressure downstream (preceding pressure rise resulting from valve closing), psi	1090
Mass flow rate through valve preceding valve closure, lb/sec	10,700

b. Pipe Break - Low Moisture Content (Outside the Containment)

Flow conditions are based on a relief area downstream of the MSIV that results in a mass flow rate approximately 300 to 400 percent of the maximum flow rate at normal operating conditions, i.e., low moisture content and high velocity.

The mass flow rate tabulated in this paragraph is a function of the flow restrictor throat diameters and the steam generator no-load pressure (reservoir pressure at initiation of transient). However, the mass flow rate to the relief opening during the first milliseconds (<250 milliseconds) of the transient (i.e., the steam contained between the flow restrictor and valve) is 2400 lb/sec per ft^2.

When this limited quantity of steam has been expelled, the flow rate decreases to the following values as a result of limitations by the flow restrictors:

Flow Conditions	
Steam generator outlet no load pressure, psia	1107
Static pressure at valve minimum flow area, atmospheric pressure downstream (preceding pressure rise as a result of valve closing), psia	1090
Mass flow rate through valve preceding valve closure, lb/sec	3350

c. Pipe Break - Reverse Flow, Low Moisture (Inside the Containment)

Initial flow through the relief opening from the piping system consists of dry steam (assumed 100-percent quality) contained in the piping system downstream of the flow restrictors.

Flow conditions are based on a large relief opening area upstream from the MSIV which results in a large reverse steam flow from the interconnected main steam lines. This reverse steam flow results in a mass flow rate approximately 10 times

the maximum flow rate at normal operating conditions, low moisture content, and greatly increased velocity.

Flow Conditions

Steam generator outlet no-load pressure, psia	1107
Static pressure at valve minimum flow area, atmospheric pressure downstream (preceding pressure rise resulting from valve closing), psia	620
Mass flow rate through valve preceding valve closure, lb/sec	11,000

d. Pipe Break - Reverse Flow, High Moisture (Inside the Containment)

After the dry steam has been exhausted, low-quality steam (4-percent steam quality) is discharged by the unfaulted steam generator(s), through its (their) associated flow restrictors and interconnected main steam piping, to the relief opening. This flow results in maximum mass flow rate and maximum moisture content; velocity is approximately three times the normal operating conditions.

Flow Conditions

Steam generator outlet no-load pressure, psia	1107
Static pressure at valve minimum flow area, atmospheric pressure downstream (preceding pressure rise resulting from valve closing), psia	680
Mass flow rate through valve preceding valve closure, lb/sec	32,100

2. Testing

The MSIVs are designed and shop tested in accordance with ASME B&PV Code, Section III.

a. Shell Hydrostatic Testing

Each valve shell is hydrostatically pressure-tested at 2250 psi in accordance with ASME B&PV Code, Section III, table NB3531-9 (for a 600 lb welded end valve).

b. Disc Hydrostatic Test

Each valve disc is hydrostatically pressure-tested at 1500 psi in accordance with ASME B&PV Code, Section III, Para NB3531.2(c). Leakage under disc hydrostatic testing shall not exceed three cc/hr/in. of valve seat diameter.

c. Valve Operator Test

There is zero leakage from all accessible static seals and joints of the system. The test is conducted when the valve has reached the full extreme in the open and closed direction.

d. Closing Rate Test

The complete valve assembly is tested to ensure that the closing time is less than five sec.

3. Leakage

The valve disc and seat materials are such that valve wear does not increase the leakage rate after a minimum of 500 cycles under normal operating conditions.

4. Design Bases

Table 10.3-4 shows the design bases of the MSIVs.

10.3.2.4 Main Steam Isolation Bypass Valves

The MSIVs are provided with 4-in. bypass valves which are normally closed. If the bypass valves were open they would tend to negate the protection provided by the MSIVs. Therefore, all four bypass valves are locked closed during power operation. During startup, hot standby and hot shutdown one MSIV bypass valve may be opened provided the other three bypass valves are locked closed and their associated MSIVs are closed. Table 10.3-5 shows the design bases for the main steam isolation bypass valves.

10.3.2.5 Flow Restrictors

Each steam generator is provided with flow restrictors which are located inside the steam generator outlet nozzle. These restrictors (several venturis arranged in a bundle) limit the steam flow rate in the event of a steam line rupture. These restrictors also minimize the thrust force effects on the steam generator and piping system.

The design basis, description, and test and inspections are included in Section 5.4.4.

10.3.2.6 Auxiliary Feedwater Pump Turbine Steam Supply

A steam supply line is provided upstream from the isolation valve (see Figure 10.3-1), located in the steam outlet line from two steam generators, to supply motive steam to the turbine drive of the auxiliary feedwater pump. This line ensures a source of steam to the turbine-driven auxiliary feedwater pump when steam generators are isolated and are producing steam from reactor

decay heat. Each line is provided with a check valve for isolation in the event of a main steam line break.

Each line is also provided with a pneumatically operated, fail-open, steam supply valve of Safety Class 2 design at its junction with the main steam line. Safety-related air accumulator tanks are provided to permit closing these valves, to satisfy containment isolation or to isolate a depressurized steam generator. A locked-closed manual valve is provided in a warm-up bypass line around each of these steam supply valves. The bypass line can be used to pre-warm the steam supply lines. The remainder of the line is constructed to Safety Class 3 requirements.

10.3.2.7 Main Steam Line Drainage

Each main steam line upstream from the MSIVs is provided with a drain pot and drain piping and valves.

The drain lines which run from the drain pot up to and including the first pneumatically operated drain valves are Safety Class 2. These valves prevent steam generator blowdown by automatic closure upon receipt of a MSIV close signal.

Each valve can be manually opened or closed from a control board switch, which is a two-train module switch. Valves fail closed on electric signal failure or air failure. Indicating lights for valve positions and a monitor light for valve closed position are located on the control board.

10.3.3 EVALUATION

The portions of main steam lines from the steam generators, out through the Containment, and up to and including the first moment restraint beyond the MSIVs, are Safety Class 2.

The safety considerations related to a main steam line break are discussed in Section 3.6. Protective measures and criteria for separation of all portions of steam lines within and outside the Containment are described in Section 3.6.5.

The seismic design is discussed in Sections 3.2 and 3.7.

The attachment of the main steam piping to the steam generators takes into account the movement of, and the allowable forces and moments on, the nozzles, as specified by the steam generator manufacturer for all operating conditions.

The environmental design bases and inservice inspection requirements are given in Subsection 10.3.1.

The valve reaction forces on the piping and the sequential blowing effect of the steam safety valves and atmospheric relief valves, located in the main steam piping, are evaluated for the design.

The interfacing of the main steam lines with the main turbine stop/control valves is determined by taking into account the movement of, and the allowable forces and moments on, the nozzles, as specified by the turbine generator manufacturer.

For the loading combinations and design stress limits relating to the safety-class main steam piping, see Section 3.9.2.

Steam is conducted from each steam generator in a separate line through the Containment Building each line is anchored at the Containment Wall. The lines have the flexibility to absorb the thermal expansion. The main steam lines run from the Containment to the Turbine Building, passing through the highest level of the Safeguards Building. Each main steam line (with its associated isolation, safety, and relief valves) is located in a compartment (connected with accesses) which provides isolation from the rest of the building and separation from the other main steam lines.

10.3.4 INSPECTION AND TESTING REQUIREMENTS

Before placing the system into service, foreign material and oxides are removed from the piping. During cleaning, entry of any fluid into the steam generators is prevented. The main steam lines are hydrostatically tested to confirm leaktightness. The testing of Class 1 and 2 components conforms to the requirements of Articles NB 6000 and NC 6000, respectively, of the ASME B&PV Code, Section III.

The hydrostatic test pressures for the valve body (discharge side) of the Safety Class 2 flanged safety valves are 1.5 times the safety valve outlet discharge pressure. Pipeline expansion and movement from the cold condition to the hot normal operating condition is checked by measuring movements from field bench marks, such as steel columns or pipe supports, as specified on design isometric piping drawings indicating calculated movements along the x, y, and z axes.

10.3.4.1 Safety Valves

The main steam safety valves are individually tested during preoperational tests. Pressure gauges mounted in the steam piping indicate the actual values of opening and closing pressures of the valves. These values are compared to the design values.

10.3.4.2 Atmospheric Relief Valves

The set point of the main steam relief valves is adjusted during preoperational tests. The valves have the capability of being stroked at any time, either manually by handwheel or remotely with manual-auto station, with their isolation valves closed.

10.3.4.3 Main Steam Isolation Valves

Periodic in-plant tests are conducted to demonstrate the capability of the MSIVs to respond to a test close signal and close within the specified time. The valve design provides for regular in-service testing of partial valve stroke. A provision is made for all solenoid-operated valves that respond to a trip signal which is exercised without interrupting availability of the trip mechanism. The preoperational and periodic testing of the MSIVs is discussed in Section 14.2.12.1.

10.3.5 WATER CHEMISTRY

A Secondary Chemistry Program will be developed for CPNPP. The objectives of the program will be to:

1. Minimize corrosion of system materials.

2. Limit the accumulation of sludge in the steam generators.

3. Minimize scale formation on the heat transfer surfaces.

4. Minimize the potential for formation of free caustic.

5. Maintain the dissolved oxygen level to optimum levels.

6. Monitor for primary to secondary leaks.

7. Maintain Steam Generator tube integrity.

8. Minimize Turbine Deposits due to carry over and volatility from the Steam Generator.

These objectives will be achieved by careful chemistry control which will include a comprehensive sampling and analysis of the secondary plant systems.

The chemistry program will entail an all volatile treatment using amines for pH adjustment and oxygen control.

The control of solids and non-volatile radioactivity in the secondary side is detailed in Section 10.4.6, Condensate Cleanup System, and Section 10.4.8, Steam Generator Blowdown System. A description of the continuous monitors is obtained from Table 10.4-20, while flow diagrams trace the location of the sampling points of the secondary system and Section 9.3.2 describes the process sampling system.

To assure the implementation and proper performance of the Secondary Chemistry Program, appropriate procedures have been developed. These procedures are contained in the Chemistry/Radiochemistry Manual and are reviewed and approved as specified by plant procedures. Procedures for this program include:

1. Identification of a sampling schedule for the critical parameters and of control points for these parameters.

2. Identification of the procedures used to measure the value of the critical parameters.

3. Identification of process sampling points.

4. A description for the recording and management of data.

5. A description of actions to be taken for off-control point chemistry conditions.

6. Identification of the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate the corrective action.

The CPNPP secondary water chemistry program developed for plant startup was based on the specification of the NSSS vendor (Reference 1), EPRI Guidelines (Reference 2), and Branch Technical Position MTEB 5-3, Revision 1 (Reference 3).

Due to the complexity of the corrosion phenomena, the understanding and control of it continues to advance along with the state-of-the-art of the methods used to measure its critical parameters. It therefore is the policy of CPNPP to make changes in its Secondary Chemistry Program that reflect additional understanding of the corrosion phenomena and the changes in the state-of-the-art of its control and critical parameter measurements. The secondary chemistry water quality will meet the specifications of the EPRI guidelines, except where CPNPP evaluations justify exceptions. Changes to the EPRI guidelines will be evaluated for applicability at CPNPP.

In the event of a steam generator tube leak, additional amine can be fed to the steam generator to counter the effect of the borated primary coolant, thus allowing proper adjustment of the pH. Operation in the basic pH range causes the disproportionation of iodine in solution to iodate and iodide. This causes a decrease in the ratio of iodine in the gaseous phase to that in solution. Steam generator blowdown is not flashed but cooled in a heat exchanger and is then demineralized, as described in Section 10.4.8. In the condenser, as in the steam generator, operation at basic pH decreases the gaseous phase iodine. Thus, only minimal amounts of iodine are expelled through the condenser vacuum pumps.

Leakage detection and monitoring of water chemistry is accomplished by the process sampling system which is described in Section 9.3.2.

Iodine partition coefficients based on the expected pH levels are 0.01 in the steam generator and 0.15 in the condenser.

10.3.6 MAIN STEAM AND FEEDWATER SYSTEM MATERIALS

The typical material specifications used in the Containment pressure boundary components are listed in Tables 10.3-6 through 10.3-9. In some cases, this list of materials may not be totally inclusive. However, the listed specifications are representative of those materials used.

10.3.6.1 Fracture Toughness

All pressure-retaining ferritic materials in the Containment pressure boundary except the feedwater isolation valves are impact-resistance tested in accordance with NC 2300 and NB 2300 of the ASME B&PV Code, Section III, mentioned in Section 5.2.4. Supplemental impact testing and fracture analysis, in conjunction with external heating of the feedwater isolation valves, are utilized to demonstrate the acceptability of the originally purchased feedwater isolation valves in lieu of impact resistance testing in accordance with the ASME B&PV Code, Division 1, Section III, Subsection NC-2300. Replacement feedwater isolation valves or their pressure retaining components, such as the valve bonnet, will be impact tested in accordance with the ASME B&PV Code, Division 1, Section III, Subsection NC-2300.

The main steam and feedwater piping are Charpy-type-impact tested in accordance with the method specified in ASME SA 370.

10.3.6.2 Materials Selection and Fabrication

The material specification used for pressure-retaining component parts in the Containment isolation boundary are in conformance with Appendix I of the ASME B&PV Code, Section III.

Where austenitic steel is used, the requirements of NRC Regulatory Guide 1.44 are followed. The insulation used with this material is in compliance with NRC Regulatory Guide 1.36. The welding procedures for this material are in compliance with NRC Regulatory Guide 1.31 to the extent specified in Section 6.1.1.1. Further discussion of fabrication, testing, and welding for austenitic steel materials is given in Section 6.1.

The cleaning and handling of all safety-related austenitic components are given in Section 6.1.

All ferritic steel components are thoroughly cleaned, descaled, and coated in accordance with the Applicant's specifications. These specifications meet or exceed the requirements of ANSI N45.2.1-1973 and NRC Regulatory Guide 1.37.

Preheat temperatures for welding low-alloy steel are in accordance with NRC Regulatory Guide 1.50. Welding is performed in accordance with NRC Regulatory Guide 1.71, where applicable.

REFERENCES

1. Westinghouse Guidelines For Secondary Water Chemistry (SGT-5.1.1 - 4468), dated February 1985.

2. EPRI PWR Secondary Water Chemistry Guidelines, Revision 2, date November 1988.

3. Standard Review Plan, Section 5.4.2.1, Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators," Revision 1.

TABLE 10.3-1
MAIN STEAM PIPE DESIGN PARAMETERS

	Nuclear-Safety Class 2 Portion[a]	Non-Nuclear-Safety Portion[a]
Outside diameter, in.	32	34
Inside diameter, in.	29.50	31.50
Minimum wall thickness, in.	1.25	1.25
Pipe material	SA 155, Class 1, Grade KCF-70	A 155, Class 1, Grade KC-70
Design pressure, psig	1200	1200
Design temperature, °F	650	650

a) For definition, see Subsection 10.3.1.

TABLE 10.3-2
MAIN STEAM SAFETY VALVES

Quantity per unit	20
Quantity per line	5
Safety class	ASME B&PV Code, Section III, Class 2, seismic Category I
Design pressure, psia	1250
Design temperature, °F	600
Accumulation	3% of set pressure
Blowdown	≤12% of set pressure
Orifice size, in^2	16

Valve Settings and Rated Flow

Set Pressure (psia)	Rated Flow For Each Valve (lb/hr)
1200	893,160
1210	900,607
1220	908,055
1230	915,502
1250	930,397

Connections	Inlet	Dual Outlet
Type	Flanged	Flanged
Size, in.	6	8 by 8
ANSI rating, lb	1500	300

TABLE 10.3-3
ATMOSPHERIC RELIEF VALVES

Quantity per unit	4	
Safety class	ASME B&PV Code, Section III Class 2, seismic Category I	
Design pressure, psia	1300	
Design temperature, °F	700	
Steam flow per valve, lb/hr		
maximum allowable	968,400 @ 1200 psia	
minimum allowable	62,150 @ 100 psia	
Normal operating pressure range, psia	965 to 1107	
Maximum operating range, psia	100 to 1300	
Normal operating temperature, °F	540 to 557	
Connections	Inlet	Outlet
Type	Butt weld	Butt weld
Size, in.	8	8
ANSI rating, lb	900	900
Actuator	Pneumatic	

TABLE 10.3-4
MAIN STEAM ISOLATION VALVES

Quantity per unit	4
Safety class	ASME B&PV Code, Section III, Class 1, seismic Category I
Design pressure, psia	1200
Design temperature, °F	600
Steam flow rate, lb/hr	4.0×10^6
Design pressure drop, psi	2.1
Normal operating pressure, psia	965 to 1107
Normal operating temperature, °F	540 to 557
Size (inlet by outlet), in.	32 by 34
Actuator, hemisphere cylinder assembly including hemisphere cylinder and transition ring	Stored energy (ASME B&PV Code, Section VIII)
Closure time, sec	5

TABLE 10.3-5
MAIN STEAM ISOLATION BYPASS VALVES

Quantity per unit	4
Safety class	ASME B&PV Code, Section III, Class 1, seismic Category I
Design pressure, psia	1200
Design temperature, °F	600
Steam flow rate, lb/hr (approximate)	60,000
Normal operating pressure range, psia	250 to 970
Normal operating temperature range, °F	401 to 541
Pressure drop	Critical
Actuator	Manual (handwheel)
Valve size, in.	4

TABLE 10.3-6
MATERIALS OF MAIN STEAM AND FEEDWATER VALVES AND PIPING

(Sheet 1 of 2)

Pneumatic/Hydraulic and Manual Gate and Check Valves	
Bodies	SA 105
Bonnets	SA 105
Discs	SA 182, F316
Closure bolting and nuts	SA 564, Type 630 and SA 194, Grade 8M
Stem	SA 564, Type 630
Air-Operated Valves	
Bodies	SA 352, Grade LCB or SA 351, Grade CF8M
Bonnets	SA 352, Grade LCB or SA 351, Grade CF8M
Discs	SA 479, Type 316L
Closure bolting and nuts	SA 193, Grade B7 and SA 194, Grade 7 or SA 453, Grade 660 and SA 194, Grade 6
Stem	SA 479, Type 316L
Miscellaneous Valves (2 in. and Smaller)	
Bodies	SA 105, or SA 182, F316
Bonnets	SA 105, or SA 182, F316
Discs	SA 479, Type 410 or SA 182, F316 or A 567, Grade 1 (Stellite No. 21)
Piping	
34 in.	A 155, Class 1, Grade KC 70
26 in. to 32 in.	SA 155 Class 1, Grade KCF 70
24 in. and smaller	SA 333, Grade 6

TABLE 10.3-6
MATERIALS OF MAIN STEAM AND FEEDWATER VALVES AND PIPING

(Sheet 2 of 2)

Fittings	
2 in. and smaller	SA 350, Grade LF 2
3 in. to 24 in.	SA 420, Grade WPL-6[a]
26 in. to 32 in.	SA 234, WPPW
34 in.	A 234, WPCW
Flanges	
4 in. to 8 in.	SA 350, Grade LF 2
34 in.	A 105

a) Due to the Unit 1 RSG installation the 18” x 16” reducing elbow material at the RSG nozzle inlet is now SA-508, Grade 2, Class 1.

TABLE 10.3-7
MATERIALS OF MAIN STEAM SAFETY VALVES

Valve body	SA 105
Inlet nozzle	SA 182, F316
Disc insert	SA 182, F316
Guide and guide ring	ACI CF8M nicology (solution annealed)
Studs (inlet flange)	SA 193, Grade B7
Nuts (inlet flange)	SA 194, Grade 2H

TABLE 10.3-8
MATERIALS OF MAIN STEAM RELIEF VALVES

Valve body	SA 216, Grade WCB
Disc insert	SA 479, Type 316
Bonnet	SA 216, Grade WCB
Bonnet studs or bolts	SA 193, Grade B7
Bonnet nuts	SA 194, Grade B7

TABLE 10.3-9
MATERIALS OF MAIN STEAM ISOLATION VALVES

Valve body	SA 216, Grade WCC
Bonnet	SA 105
Bonnet studs	SA 540, Grade B23CL4
Bonnet nuts	SA 194, Grade 7
Gland studs	A 193, Grade B7
Gland nuts	A 194, Grade 2
Disc	SA 182, Grade F11

TABLE 10.3-10
THIS TABLE HAS BEEN DELETED.

TABLE 10.3-11
BRANCH OFF STEAM FLOW PATH BETWEEN MAIN STEAM ISOLATION VALVE AND TURBINE STOP VALVE

SYSTEM IDENTIFICATION	MAXIMUM STEAM FLOW, LB/HR	TYPE OF SHUT-OFF VALVE	SIZE OF VALVE, INCH	QUALITY OF VALVE	DESIGN CODE	CLOSURE TIME OF VALVE	ACTUATION MECHANISM	POWER SOURCE	MOTIVE REMARKS
Secondary Sampling	150	Globe	3/8	Commercial	Mfg. Standard	N. A.	Manual	N. A.	Notes 1, 5
Main Steam Line	N/A	Globe	2	Commercial	Mfg. Standard	N. A.	Manual	N. A.	Notes 1, 2, 5
Condensate Removal	13,000	Globe	2	Commercial	Mfg. Standard	N. A.	Air Diaphragm	Air	Notes 1, 3
Main Steam Strainer	N/A	Globe	2	Commercial	Mfg. Standard	N. A.	Manual	N. A.	Notes 1, 2, 5
Condensate Removal	13,000	Globe	2	Commercial	Mfg. Standard	N. A.	Air Diaphragm	Air	Notes 1, 3
Steam Dump Header	N/A	Globe	2	Commercial	Mfg. Standard	N. A.	Manual	N. A.	Notes 1, 2, 5
Condensate Removal	13,000	Globe	2	Commercial	Mfg. Standard	N. A.	Air Diaphragm	Air	Notes 1, 3
Steam Dump Valve	1,109,000	Globe	8	Commercial	ANSI B16.5	5 sec.	Air Diaphragm	Air	Notes 1, 3
Steam Dump Valve	N/A	Globe	1	Commercial	Mfg. Standard	N. A.	Manual	N. A.	Notes 1, 2, 5
Line Condensate Removal	6,000	Globe	1	Commercial	Mfg. Standard	N. A.	Air Diaphragm	Air	Notes 1, 3
Feedpump Turbine	N/A	Globe	2	Commercial	Mfg. Standard	N. A.	Manual	N. A.	Notes 1, 2, 5
Steam Supply Condensate Removal	13,000	Globe	2	Commercial	Mfg. Standard	N. A.	Air Diaphragm	Air	Notes 1, 3
Auxiliary Steam Supply	10,000	Gate	3	Commercial	ANSI B16.5	15 sec.	Motor	Elect. (Not safe-guards)	Notes 1, 4, 7
Auxiliary Steam Line	N/A	Globe	2	Commercial	Mfg. Standard	N. A.	Manual	N. A.	Notes 1, 2, 5
Condensate Removal		Globe	2	Commercial	Mfg. Standard	N. A.	Air Diaphragm	Air	Notes 1, 6

NOTES:
1. Will not affect reactor shutdown.
2. Steam traps provided downstream of shut-off valves.
3. Manual closure from Control Room, normally closed, fail closed.
4. Flow is intermittent.
5. Manual closure, normally open.
6. Manual closure from the control room, normally closed, fail open.
7. Normally open, closure from the control room.

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

10.4.1 MAIN AND AUXILIARY CONDENSERS

Each CPNPP unit is equipped with one main condenser and two auxiliary condensers. The main condensers are described in Subsection 10.4.1.1 and the auxiliary condensers are described in Subsection 10.4.1.2.

10.4.1.1 Main Condensers

The twin-shell main condenser furnishes the heat sink for the main turbine exhaust steam (as discussed in Subsection 10.4.5) and for the steam dump system (as discussed in Subsection 10.4.4). The hot well also provides condensate storage for plant operations. The exhaust steam side of the main condenser is shown on Figures 10.4-8 and 10.4-9 and described in Subsection 10.4.7.

10.4.1.1.1 Design Bases

Design is based on the following conditions (nominal) at full load of the main turbine:

Conditions	Value
Exhaust steam, lb/hr	8.44×10^6 (approx.)
Condenser duty, Btu/hr	8.06×10^9 (approx.)
Condenser pressure at turbine exhaust, in. Hg abs.	3.55
Circulating water data Flow, gpm	1.03×10^6
Number of passes per shell	One
Inlet temperature, °F	95
Outlet temperature, °F	110.8

The condensers function effectively at all loads below maximum guaranteed turbine load. The main condenser is also designed to perform the following functions:

1. To condense steam released to the condenser by the steam dump system (Subsection 10.4.4) during unit startup, normal cooldown, and emergency cooldown

2. To condense turbine exhaust steam and dump steam under larger load reductions than those defined in Section 10.2.1

3. To deaerate the condensate

4. To accept drains from the feedwater heaters and other miscellaneous sources (see Subsection 10.4.11, Figure 10.4-14).

The condenser hot well provides approximately 13,800 ft^3 of condensate storage capacity, which is equivalent to approximately five-minutes operation at maximum load. Automatic rejection of hot well inventory to a volume equivalent to about 4 minutes operation at maximum load is provided. Hot well inventory is typically rejected to a volume equivalent to 3 minutes operation at maximum load during normal operation. Hot well volume equivalent to two minutes operation at maximum load is sufficient to achieve feedwater isolation and auxiliary feedwater initiation.

Following a 50-percent load rejection, the condenser accepts at least 40 percent of the main steam flow through the steam dump system. This condition is accommodated without increasing the condenser back pressure to the turbine trip set point or exceeding the allowable turbine exhaust temperature.

10.4.1.1.2 System Description

The surface condenser is of the deaerating, single pass, fixed-bottom, twin-shell, single-pressure, horizontal surface, divided water box, floor-supported type, and is sized to condense exhaust steam from the main turbine under full load conditions.

It is connected to the exhaust openings of each low-pressure turbine by an expansion joint. An equalizing line between the shells limit turbine exhaust temperature differences.

Erosion protection is provided by installation of baffles and spray headers to prevent direct impingement of steam and/or flashing mixtures which enter the shell at various locations. Condenser materials were specifically chosen to resist the effects of erosion. The titanium used in the condenser tubes has good erosion and steam impingement resistance.

The condenser is cooled by the Circulating Water System (described in Section 10.4.5), which removes the heat rejected to the condenser.

Layout drawings (Figures 10.4-1 and 10.4-2) show the connections to the turbine exhaust and the Circulating Water System. Valves are provided in the Circulating Water System to permit either half of each condenser shell to be removed from service.

Condensate from the auxiliary (steam generator feedwater pump turbine) condensers is drained to the main condenser hot well.

10.4.1.1.3 Safety Evaluation

1. Radioactive Contaminants

   The radioactivity in the main condensers is a function of the percentage of defective fuel cladding, the escape rate coefficients, the primary-to-secondary leak rate, the secondary system flow rate, the blowdown rate, and the steam generator and condenser partition factors. Steam generator tube leakage effects on the secondary system are discussed in Section 10.2.4. The effects of short term exposure of low radioactivity levels on the Condensate Cleanup System are discussed in Subsection 10.4.6.

   Steam generator blowdown processing of radioactive primary-to secondary system leakage is discussed in Subsection 10.4.8.

Radioactive contaminants from steam generator primary-to-secondary coolant leakage reach the main condenser shell through the steam path (i.e., main steam lines-turbine condenser and steam dump lines-condenser) and the liquid path (steam generator blowdown processing system).

The assumptions (partition factors, decontamination factors, and leak rates) for calculating the expected radioactivity concentrations in the Steam and Power Conversion System generally and in the steam generators, turbine and condenser specifically are discussed and tabulated in Section 11.1.

2. Vacuum Loss

There is no direct influence of the condenser operation on the Reactor Coolant System (RCS). Partial loss of condenser vacuum is annunciated in the Control Room. Should vacuum continue to decay, the turbine automatically trips and the steam dump valves to the condenser open automatically if partial vacuum can be maintained.

Following a turbine trip, the turbine trip steam dump controller becomes active. A description of the turbine trip steam dump controller is presented in Section 7.7.1. If the operation of the dump system is blocked by low condenser vacuum, heat removal from the RCS is provided by the main steam safety valves.

3. Air Leakage

The condensers are designed to minimize air leakage. Welded construction is used for the condenser shell and, wherever practicable, for condenser shell connections and penetrations.

Equipment and piping connected to the condenser shell are also designed to minimize air leakage to the condensers through the use of water seals, pressure seals, and leaktight valves. The maximum expected air leakage into the main condensers is anticipated to be 200 lb/hr during normal operation (see Subsection 10.4.2).

4. Safety-Related Equipment

The condensers are not safety-related and there is no safety related equipment in the area of the condenser which can be flooded as a result of a failure of the condenser.

5. Hydrogen Buildup

The potential for hydrogen buildup in the condenser is negligible since there are no potential sources of hydrogen in direct contact with the condenser. Any small amounts of hydrogen originating in the RCS and reaching the condenser through steam generator tube leakage are removed by the condenser evacuation system. Experience from similar plants shows that the concentration of hydrogen is well below any hazardous value.

6. Control and Detection of Condenser Leakage

The main condenser tube bundles and tube sheet assemblies are constructed of titanium and titanium clad carbon steel respectively. The main condenser tubes are rolled into and

welded to the tubesheets. This design is virtually leak tight, but should a tube-to-tubesheet leak develop, it will be collected in salinity troughs.

Tube leaks are detected by condensate conductivity meters and by installed sodium analyzers at the condenser hotwell. In the event of any leaks into the condenser, the full flow condensate polisher can reduce the dissolved and suspended solid contaminants introduced through the condenser leakage. The cation conductivity measurement on the inlet and outlet headers of the Condensate Cleanup System, as well as at the effluent of each vessel, monitor condensate quality and individual vessel performance. Section 10.4.16, Table 10.4-20, Secondary Plant Sampling System Measured Parameters, and Section 10.4.6.5, para. 3, Conductivity Instrumentation for the Condensate Cleanup System, indicate the measures taken for detecting condenser cooling water leakage into the condensate stream.

10.4.1.1.4 Inspection and Testing Requirements

Each condenser shell receives a field hydrostatic test prior to initial operation which consists of filling the condenser shell with water and, with the resulting static head maintained, inspecting all welds and surfaces for visible leakage, or excessive deflection, or both.

Each condenser water box receives a field hydrostatic test, with all joints and external surfaces being inspected.

After the completion of the tubing installation in the condenser, all tube joints are leak tested.

10.4.1.1.5 Instrumentation Requirements

1. Hot Well

   Each condenser shell is provided with local and remote hot well level and pressure indication. The remote indication is by means of indicators and alarms in the Control Room. The condensate level in the condenser hot well is maintained within proper limits by automatic controls which provide for transfer of condensate to and from the condensate storage tank as needed to satisfy the requirements of the steam system. Condensate temperature is measured in the suction lines of the condensate pumps.

2. Exhaust Hood

   Excessive temperature in the turbine exhaust hood causes a thermostatically controlled spray valve to open, thereby preventing further increases in temperature. The valve closes again once a limit for shutting off the spray is met. These limits include settings for minimum turbine speed, maximum exhaust hood temperature, and minimum steam pressure at low-pressure casing inlet. Rupture diaphragms are provided to prevent excessive overpressure of the main condenser shells. There is no safety-related instrumentation in this section.

3. Water Box

   Water box pressure and temperature measurements are provided.

4. Radioactivity

A radiation monitor is provided on the condenser vacuum pumps discharge line to detect the presence of noncondensible radioactive gases in the system and initiates an alarm upon increasing radiation levels (see Section 9.3.2 and 11.5). A radiation monitor on the steam generator process sample line header detects leakage across the tubes in the steam generator (liquid phase) and provides backup information to the condenser off-gas monitor. Turbine drains effluent is monitored for radioactivity prior to release to the low volume waste pond. Indication and alarm is initiated in the Control Room upon detection of radioactivity in excess of established limits. (See Section 11.5.)

10.4.1.2 Auxiliary Condensers

10.4.1.2.1 Design Bases

The auxiliary condensers provide the heat sink for the steam generator feedwater pump turbine exhaust steam. One auxiliary condenser is provided for each steam generator feedwater pump turbine. The condensate flows into the main condenser hot well.

10.4.1.2.2 System Description

Each auxiliary condenser is of the single-pass, nondivided water box, floor-supported type, and is sized to condense exhaust steam from the steam generator feedwater pump turbine under full-load conditions. Condensate from the auxiliary condenser hot wells is drained to the main condenser hot well. Each condenser is connected to the exhaust openings of the turbine by an expansion joint. The auxiliary condenser is cooled by the Circulating Water System (described in Subsection 10.4.5) which removes the heat rejected to the condenser.

An atmospheric relief valve is provided for each condenser which discharges to the atmosphere through a 20 in. vent connection. No provision is made for the monitoring of radioactivity in the effluent from these relief valves since expected potential radioactive concentrations and flow rates provide effluents well below established limits.

10.4.1.2.3 Safety Evaluation

Safety evaluation considerations for the auxiliary condensers are the same as those for the main condensers as described in Subsection 10.4.1.1.3.

10.4.1.2.4 Inspection and Testing Requirements

See Subsection 10.4.1.1.4.

10.4.1.2.5 Instrumentation Requirements

Each condenser shell is provided with a pressure transmitter for remote indication in the Control Room. The condenser hot well is provided with local level indication. The condensate level in the hot well is monitored by means of high water level alarm in the Control Room. A high turbine exhaust hood temperature alarm is provided in the Control Room.

Steam generator feed pump turbine trip is activated on loss of auxiliary condenser vacuum, with condenser back pressure reaching or exceeding a set point of between 12 and 13 in. Hg abs. A high back pressure alarm setting is in the range of 8.5 to 9.5 in. Hg abs.

Water box pressure and temperature measurements are provided.

There is no safety-related instrumentation in this section.

10.4.2 CONDENSER EVACUATION SYSTEM

The Condenser Evacuation System, as shown on Figure 10.4-3, removes air and noncondensible gases from the main and auxiliary condensers, and it removes trapped air and primes those parts of the Circulating Water System which are above lake level.

This system is considered non-nuclear-safety-related in accordance with ANSI N18.2 and is designated non-seismic Category I.

10.4.2.1 Design Bases

The Condenser Evacuation System is designed for initial evacuation of the main condenser shells and the two auxiliary condenser shells at startup, and for removal of noncondensible gases during normal plant operation.

A separate priming portion of this system primes the Circulating Water System by removing air from the (a) high points in the circulating system piping, (b) main and auxiliary condenser water boxes, and (c) the turbine plant cooling water heat exchanger and the condenser vacuum pump heat exchanger water boxes.

As an alternate means of initiating turbine trips, the system incorporates a vacuum breaker arrangement for the main and auxiliary condenser shells. A vacuum breaker is also provided to protect the circulating water system from system transients.

The system is designed to prevent uncontrolled release of radioactive material to the environment, in accordance with 10 CFR Part 50, Appendix I, and General Design Criteria (GDC) 60 and 64. A radiation monitor is used to provide detection of radioactivity.

The capacity of each vacuum pump is as follows:

Hogging capacity, acfm dry air, at 10 in. Hg abs. is 2300. Design holding capacity, scfm, at 1.0 in. Hg abs. is 25.

10.4.2.2 System Description

10.4.2.2.1 System Operation

For each unit of CPNPP, three 100-percent capacity, motor-driven, two-stage, rotary-type condenser exhausting vacuum (CEV) pumps are furnished for the Condenser Evacuation System. These pumps provide hogging and holding functions for the main and auxiliary condenser shells.

The first function of the CEV pumps is the initial evacuation of the main and auxiliary condenser shells prior to the availability of the steam during plant startup. This operation, called hogging, uses the three CEV pumps operating in parallel. The pump suction lines are connected via a common suction header with the main and auxiliary condenser shells. When started, the three CEV pumps provide for the evacuation of air from the condenser shells at a rate of 2300 acfm dry air per pump at approximately 10 in. Hg absolute. The hogging process brings the condenser shell pressures down to 10 in. Hg in approximately one hour.

At approximately 10 in. Hg abs., the CEV pumps automatically switch to the holding mode. If for some reason the condenser shell pressures increase beyond this setting during the holding operation, the CEV pumps automatically revert to the hogging operation.

During the hogging operation, the volume of air handled by the first stage is greater than the inlet capacity of the second stage of each CEV pump. The CEV pumps use an internal arrangement in which the exhaust bypasses the second stage and is separated and finally discharged. At approximately 10 in. Hg abs., the pumps automatically switch from the hogging to the holding mode, and the second stage handles the entire first-stage discharge. The internal bypass arrangement, now closed, allows the two-stage combination to pull the system down to operating pressure. The second-stage exhaust is then separated and discharged.

During the holding sequence, the number of CEV pumps in operation can be reduced. At any load during the winter, with the circulating water inlet temperature at 40°F, two CEV pumps are required to maintain the condenser vacuum at approximately one in. Hg absolute. One CEV pump is sufficient to maintain a condenser pressure of 3.5 in. Hg abs. at any load with 95°F inlet water.

Seal water makeup for the CEV seals is provided by the demineralized and reactor makeup water system. The circulating water system cools the CEV pump seal water in the CEV pump heat exchangers. The cooled seal water is then recirculated through the CEV pumps. Water vapor pulled from the condenser shell which comes into contact with the CEV pump seal water is condensed. Water from the CEV pump seals drains continuously by gravity to Turbine Building Local Floor Drain, which eventually is disposed of by the liquid waste processing system.

A vacuum breaker system is provided for the main and auxiliary condenser shells. This system consists of piping and motor-operated valves which, when actuated, allow air to enter the condensers at a rate in excess of the vacuum pump capability to remove air. This results in high back-pressure in the condenser and initiates a turbine trip, thereby providing another method of turbine trip, should, for any reason, the normal method become inoperative.

For each unit of CPNPP, two 100 percent capacity, motor driven, single stage rotary-type condenser water box vacuum priming pumps provide for priming of the Circulating Water System, and the water boxes of the Main Condenser, the Auxiliary Condensers, the Turbine Plant Cooling Water System heat exchanger, and the seal water heat exchanger of each CEV pump.

The priming operation removes air from the Circulating Water System and several condenser and heat exchanger waterboxes. The partial vacuum causes them to fill with water from Squaw Creek reservoir. During normal operation, the condenser water box vacuum priming pumps continue to remove accumulated air at high points in the Circulating Water System and the several water boxes.

Priming valves and standpipes are provided to prevent carryover of the potentially corrosive circulating water to the condenser water box vacuum priming pumps.

Condenser water box vacuum priming pump seal water is discharged to the Turbine Building Local Floor Drain. Seal water for the condenser water box vacuum priming pumps is provided by the Demineralized and Reactor Makeup Water System.

A vacuum breaker is provided on the main condenser outlet water boxes. The vacuum breaker is designed to protect the tube side components from transients following a simultaneous trip of all four circulating water pumps. This system is comprised of piping and an air operated valve which will open on loss of all four circulating water pumps.

In order to minimize air inleakage at the main condenser shell and water box vacuum breakers, and the auxiliary condenser relief valves and vacuum breakers, these components are sealed by water from the Condensate System.

Safe shutdown of the unit does not rely on the availability of the Condenser Evacuation System. Hence, there is no provision to provide power to this system from the standby diesel generators.

The loads in this system are supplied with power from a non-class 1E power supply.

10.4.2.2.2 Equipment Design Criteria

Design criteria for equipment are listed in Table 10.4-1.

10.4.2.3 Safety Evaluation

The safety evaluation of this system is given in Subsection 10.4.1.1.3, in conjunction with the evaluation of the condensers. The radiological considerations discussed therein result in the maximum anticipated radioactive Containment discharge rates for this system given in Table 11.2-10.

The off-gas CEV pump discharge is potentially radioactive; therefore, if steam generator tube leakage occurs coincident with fuel cladding failures, the CEV pump discharge gases are cascaded into a common discharge line which is directed to the heating, ventilating, and air-conditioning (HVAC) system for controlled release to the atmosphere through an exhaust unit, which consists of roughing, HEPA, and charcoal filters and the plant vent.

A radiation monitor is provided on the condenser vacuum pumps discharge line to continuously measure the radiation level of the condenser off-gases discharged to the atmosphere with indication given in the Control Room. A sample pump diverts a small percentage of the discharge flow through the monitor and back to the discharge line.

Radioactive elements present in the secondary system which could ultimately reach the condenser off-gas can also be detected by the steam generator liquid sample monitor (Section 11.5).

The presence of radiation above a set point is annunciated in the Control Room where, upon receiving this alarm, a manual sample of the off-gas is drawn to determine if the condensers are the source of the activity.

10.4.2.4 Tests and Inspections

All tests and inspections of equipment in the Condenser Evacuation System are performed in accordance with applicable codes.

10.4.2.5 Instrumentation Requirements

Following a turbine trip, the turbine trip steam dump controller becomes active. A description of the turbine trip steam dump controller is presented in Section 7.7.1. If the operation of the dump system is blocked by low condenser vacuum, heat removal from the RCS is provided by the main steam system safety valves.

The Condenser Evacuation System is provided with the following instrumentation to monitor system performance:

1. The suction header of the CEV pumps contains a pressure switch to alarm on low vacuum and a pressure gauge for local indication.

2. The suction header of the water box priming pumps contains a pressure gauge for local indication.

3. The standpipe that prevents water carryover into the condenser water box vacuum priming pump suction header contains a pressure switch that on low vacuum primary header pressure provides input to a main condenser priming trouble alarm.

4. The main condenser vacuum priming trouble alarm is also activated if the main condenser outlet waterbox vacuum breaker air operated valve is not fully closed.

The CEV pumps are controlled from the Control Room. Indicating lights in the Control Room advise operating personnel of the pump or pumps in operation and of the position of the pump suction valves. Motor trips for the CEV pumps are annunciated in the Control Room.

Motor-operated vacuum breaker valves are provided with Control Room mounted control switches and indicating lights.

A local pressure gauge is provided for monitoring the pressure of the CEV Pump seal water makeup supply from the Demineralized and Reactor Makeup Water System.

The condenser water box vacuum priming pumps are controlled from a locally mounted control panel. Indicating lights on the panel advise operating personnel of the pump(s) in operation.

10.4.3 TURBINE GLAND SEALING SYSTEM

The Turbine Gland Sealing System, shown on Figure 10.4-4, is designed to prevent an inflow of air into the main and auxiliary condensers via the shafts of the main turbine and the feedwater pump turbine drivers, respectively. The system is also designed to prevent an outflow of steam from the high-pressure and low-pressure turbine shafts. System components are located in the Turbine Building.

The system is designed as non-nuclear-safety-related in accordance with ANSI N18.2 and is classified as non-seismic Category I [3], [4]. The system is designed with provisions to monitor and control releases of gaseous radioactive material in accordance with GDC 60 and 64 of 10 CFR Part 50 [1], [2].

10.4.3.1 Design Bases

1. Main Turbine Seals

The main turbine is self-sealing at loads above approximately 40 percent. The seal steam supply header has a design flow rate of 14,370 lb/hr at 14.83 psia. At loads below approximately 40 percent, auxiliary steam from the Auxiliary Steam System is supplied at 6832 lb/hr, 164.7 psia, and 368°F.

Design conditions for dumping excess steam to feedwater heaters 6A and 6B are 13,154 lb/hr at 4.43 psia and 3.4-percent moisture. The gland steam condenser and exhauster are designed to maintain one to two in. water vacuum in the seal steam condenser header while condensing 1670 lb/hr of steam and passing 1050 lb/hr of air through the plant radiation monitor and exhaust plenum. The gland steam condenser tubes are designed for 1140 gpm of condensate at a design temperature and pressure of 125°F and 600 psig, respectively.

2. Feedwater Pump Turbine Seals

The seal steam supply header has a normal flow rate of 230 lb/hr at 2 to 4 psig.

At startup, 1150 lb/hr of auxiliary steam from the Auxiliary Steam System is supplied at 2 to 4 psig. During an emergency, seal steam is supplied from the main steam supply system.

The auxiliary gland steam condenser and exhauster are designed to maintain 7- to 10-in. water vacuum in the seal steam condenser header while condensing 780 lb/hr of steam and passing 170 lb/hr of air through the plant radiation monitor and exhaust plenum.

10.4.3.2 System Description

1. Normal Operation Conditions

a. Main Turbine Steam Seals

Both high-pressure and low-pressure turbines have the same physical seal arrangement. At every point where the turbine shaft penetrates the turbine casing, there are two annular chambers separated by labyrinth seals from each other, the atmosphere, and the interior of the turbine.

The high-pressure turbine seals provide seal steam for the low-pressure turbines and prevent the loss of steam to the atmosphere. The high-pressure steam leaks through the labyrinth seals and is bled off from the inner annular chamber to supply steam to the low-pressure turbine seals at 14.83 psia (slightly above atmospheric pressure). The steam that is not bled off is drawn into the outer

annular chamber along with air from the atmosphere. The outer annular chamber is kept at a slight vacuum (1- to 2-in. water vacuum) by the main turbine gland steam condenser.

The low-pressure turbine seals prevent air from entering the turbine and steam from escaping to the atmosphere. Steam (14.83-psia) from the high pressure turbine enters the inner annular chamber and flows both inward and outward along the shaft. The steam that flows outward joins air flowing inward at the outer annular chamber, where it is drawn into the main turbine gland steam condenser.

The main turbine gland steam condenser is kept at a vacuum by the main turbine gland steam condenser exhauster during all modes of turbine gland steam system operation. Air and steam from the outer annular chambers of all main turbines and from the main steam stop and control valve leakoffs is drawn into the main turbine gland steam condenser, where the steam is condensed by cool condensate and drained through a loop seal to the atmospheric drain tank.

Air, noncondensable gases, and some water vapor are ejected by the main turbine gland steam condenser exhauster. The water drains to the atmospheric drain tank through a loop seal, and the air and noncondensable gases are vented to the primary plant ventilation exhaust system, which consists of 16 exhaust filtration units. See Section 9.4 for a description of the exhaust filtration units. Furthermore, the exhaust is monitored for radiation prior to release to the atmosphere.

During startup, shutdown (condenser at partial vacuum), and when turbine load falls below approximately 40 percent, high-pressure turbine seal pressure is insufficient to provide seal steam for the low-pressure turbines. Under these conditions, all main turbine seals operate in the same manner as the low-pressure turbine seals during normal operation. Steam (14.83 psia) is supplied by the Auxiliary Steam System through a pressure-regulating steam supply valve to the inner annular chamber of both the high-pressure and low-pressure turbines. A warmup line is provided upstream from the supply valve. Any overpressure of the seal steam header is relieved through the 10 in. leakoff valve to feedwater heaters 6A and 6B. The feedwater heater tube bundles are protected from thermal stress by an impingement plate and are suitable for this steam temperature.

Piping that may contain condensate during normal operation collect water in drain pots which discharge to the main condenser. There is also a drain pot on the seal steam header which drains to the Turbine Building Drain System through a motor operated valve.

b. Feedwater Pump Turbine Steam Seals

Both high-pressure and low-pressure seals of the feedwater pump turbine have the same seal arrangement as those for the main turbine.

Steam which is slightly above atmospheric pressure (2 to 4 psig) is piped from the high-pressure inner annular chamber to the low-pressure inner annular chamber to prevent air inleakage at the low-pressure end of the turbine which is at

condenser vacuum. The outer annular chambers at both ends of the turbine are held at a slight vacuum (7- to 10- in. water vacuum). Therefore, at the high-pressure end, most of the steam is bled off to seal the low-pressure end, and the rest of the steam plus air is drawn into the auxiliary gland steam condenser. At the low-pressure end, seal steam is supplied at three psig to prevent the ingress of air, and steam plus air is drawn into the auxiliary gland steam condenser to prevent the escape of steam to the atmosphere.

The pressure at the high-pressure end is insufficient to seal the low-pressure seals. Auxiliary steam from the Auxiliary Steam System is provided at the inner annular chambers through the steam seal supply valve.

During all modes of plant operation, the auxiliary gland steam condenser applies a vacuum to the outer annular chambers of the feedwater pump turbine seals. Vacuum is maintained by the auxiliary gland steam condenser exhauster which dumps air and noncondensable gases to the Primary Plant Ventilation Exhaust System and drains water to the atmospheric drain tank through a loop seal. Steam is condensed by cool condensate, and the condensed steam drains to the atmospheric drain tank through a loop seal to maintain condenser vacuum.

Piping that may contain condensate during normal operation collects water in drain pots which discharge to the main condenser.

2. Abnormal Conditions

a. Main Turbine Steam Seal System

Upon high water level due to the tube break in the main turbine gland steam condenser, a level switch actuates an alarm to indicate the condition. If the water backs up further, it enters an emergency overflow connection, which provides emergency drains to the atmospheric drain tank. No valves are required in the emergency overflow connection because of a loop seal which maintains vacuum during normal operation. The loop seal is initially filled from the condensate system.

Upon loss of the main turbine gland steam condenser exhauster, the system is capable of continued operation for a limited time by venting steam collected from the outer annular chambers to the atmosphere through the emergency bypass to the atmosphere. In this mode of operation, air does not leak into the low-pressure turbines, but steam leaks out of all turbine seals to the Turbine Building.

In the event of primary to secondary leakage in the steam generators, there will be very low releases of radioactivity when the main turbine gland steam condenser is being bypassed.

b. Feedwater Pump Turbine Steam Seal System

In the event of excessive packing wear or blowout, the packing leakage increases drastically; this causes the steam seal supply valve to modulate to maintain sealing pressure of four psig. As the SSSV controller senses a rising gland steam

pressure, its pneumatic output increases to throttle close the supply valve. If the steam seal supply valve cannot maintain pressure at four psig, the pressure increases until the relief valve starts opening at about 20 psig. At 25-psig seal header pressure, the relief valve is wide open and further increase in flow increases header pressure beyond 25 psig.

Loss of air supply to the controller causes the steam seal supply valve to open wide, and any pressure increase beyond 20 psig again opens the relief valve.

Main unit trip reduces the pump turbine load to the point where the steam seal supply valve opens to maintain the seal header pressure at the set point value.

3. Electrical System

Safe shutdown of the unit does not rely on the availability of the Turbine Gland Sealing System. Therefore, there is no provision to provide power to this system from the standby diesel generators.

The loads in this system are supplied power from the non-Class 1E power supply.

10.4.3.3 Safety Evaluation

The amount of noncondensable gases released by the condensed steam is negligible (as shown in Table 11.1-1), and there is no need for an activity monitor at the gland seal condenser exhauster. However, air and noncondensable gases are exhausted to the primary plant exhaust plenum, which is monitored and consists of a series of filtration units.

For more details, refer to Section 11.3.

10.4.3.4 Tests and Inspection

Normal preoperational tests are performed in accordance with applicable codes. Operability of the gland seal system is verified by the maintenance of condenser vacuum and visual inspection of the equipment is performed before and during operation.

10.4.3.5 Instrumentation Requirements

1. Main Turbine Steam Seals

Gland steam header pressure is maintained at a reference value by an automatic control system supplied by the turbine-generator manufacturer. The seal steam supply valve and seal steam leakoff valve are automatically modulated to maintain header pressure.

The valves can also be controlled manually from Operating and Monitoring (OM) screens at computer workstations in the Control Room.

The seal steam condenser exhauster fan is started and stopped manually with a Control Room switch. Water accumulation in the seal steam header is alarmed in the Control Room by a level switch. The operator then manually opens the motor-operated valve.

Seal steam header pressure indication is provided locally and remotely on the OM screens.

The following system abnormalities are alarmed in the Control Room:

a. Seal steam condenser level high

b. Seal steam control valve failure

c. High drip pot levels

d. Seal steam exhaust fan trip

e. Drain seal steam header level high

Forward-reverse pitot tubes are installed in all seal steam supply lines (both main turbines and feed pump turbines). These can be used to measure abnormal gland wear. Temperature test wells are also installed in these lines. A forward reverse pitot tube and a test temperature well are located in the leakoff steam line to the extraction steam system.

2. Feedwater Pump Turbine Steam Seals

Gland steam header pressure is maintained at a three psig set point by modulation of the steam seal supply valve. A split range controller sets the air pressure at the control valve so that sufficient steam is admitted from the auxiliary Steam System (if needed).

Seal steam header pressure indication is provided locally and remotely in the Control Room.

The following system abnormalities are alarmed in the Control Room:

a. High auxiliary gland steam condenser level

b. Low seal steam supply header pressure

c. High differential pressure across the control valve air supply filter

d. High drip pot level

10.4.4 STEAM DUMP SYSTEM

10.4.4.1 Design Bases

To enable the NSSS to follow turbine load reductions which may exceed a 10-percent step or five percent per minute ramp (Section 10.2.1), the capability of creating an artificial steam load is incorporated in the Steam and Power Conversion System. This load is created by dumping steam directly to the main condensers, thus bypassing the turbine.

Each unit of the CPNPP is provided with a steam dump system rated at greater than 40 percent of the steam flow at the ESF rating nominally reflected on the Valves Wide Open Heat Balance, Figure 10.1-2. This flow capacity permits the turbine to take a 50-percent load reduction without reactor trip. This system also allows a turbine and reactor trip to occur from full load without lifting the steam generator safety valves.

During normal reactor plant cooldown, the steam dump system is used to control cooldown rate by bleeding steam from the steam generators to the condensers.

Table 10.4-2 shows the design bases of the steam dump system.

The steam dump lines are designed in accordance with ANSI B31.1, Power Piping Code.

10.4.4.2 System Description

10.4.4.2.1 General

The steam dump system is shown on Figure 10.3-1 (Sheets 1 and 2).

Steam is discharged to the condensers through 12 air-operated, partial capacity, automatically controlled steam dump valves, which are installed in the steam dump lines located in the Main Steam Supply System between the steam generator isolation valves and turbine stop and control valves.

The steam dump valves are arranged in parallel so that, when combined, they permit the desired bypass flow to pass. This arrangement limits the steam bypassed to the condenser should a valve open accidentally or stick open, thereby minimizing the potential hazard of an uncontrolled cooldown rate of the primary system. This arrangement also permits the steam dump flow to be evenly shared by the turbine condensers, thus preventing uneven turbine exhaust back pressures.

The steam dump system is designed to bypass steam around the main turbine to the condenser to accomplish the following:

1. To permit a generator load rejection of up to 50 percent from maximum guaranteed power level without a reactor trip

2. To prevent lifting of the main steam safety and power-operated relief valves following a generator load rejection of up to 50 percent from the maximum guaranteed capability power level

3. To discharge to the condenser the steam generated during plant cooldown at a rate that limits thermal transients to allowable values; steam generator depressurization would continue in this mode down to a level in which the plant cooldown process is transferred to the RHR System

4. To maintain steam system header pressure at required pressure during startup, hot standby, and reactor physics testing periods

10.4.4.2.2 Steam Dump Valves

Twelve steam dump valves are provided for each unit, with each valve having a capacity of greater then 530,000 lb/hr. They are designed to fail closed on loss of either air or control signal and are split into two groups to ensure even distribution of heat into each condenser. These valves are arranged for automatic operation and for remote manual control from the control room. The dump valves are provided with local and remote open/closed position indicators. The dump valves also are capable of fast actuation or of operating on throttling service.

The steam dump valve actuation characteristics are as follows:

1. The valves are capable of going from full closed to required flow within three sec after receiving a trip/open signal. This includes the time required to actuate the solenoid valves associated with each dump valve.

2. The valves are capable of going from full open to full closed in five sec after de-energization of the solenoid valves.

3. The valves are capable of being modulated with a maximum full stroke time of 20 seconds to achieve required flow.

4. To ensure adequate dump control at a hot shutdown condition or during cooldown, the steam dump system is designed to control the total dump flows down to 80,000 lb/hr over the steam pressure in the range of 100 to 1300 psia.

5. The dump valves can be actuated either by the reactor coolant average temperature control logic or by the steam system header pressure depending on a selection made by the operator in the control room.

10.4.4.2.3 System Operation

During operating transients for which the plant is designed, the steam dump system is automatically regulated in the average reactor coolant temperature (Tavg) control mode to maintain the programmed Tavg.

A programmed reactor coolant reference temperature (Tavg) corresponds to each turbine load. During load variations when the reactor and turbine outputs are unbalanced, it deviates from the actual Tavg of the primary coolant. The magnitude and the rate of this deviation, which depends on the transients, provides a signal that selects a new control rod pattern and activates the adequate number and mode of operation for the dump valves.

During a load reduction, the valves are modulated by temperature deviation through a load rejection controller. When the mode selection switch is in the steam pressure, the valves are modulated to maintain the steam header pressure setpoint.

On large step load reductions (above 10 percent) or reactor trip, the steam dump valves open rapidly, in three seconds or less. During the three second period, while the turbine valves are closing and dump valves are opening, there is a temperature rise in the NSSS and an approximately 100-psi pressure rise in the steam generators. In the initial part of a large step load transient, all dump valves go fully open. Then the valves are modulated closed in sequence

to obtain a design load change in the reactor of five percent per minute. The valves are fully closed when the reactor power matches the turbine power.

Reactor trip transfers steam dump control from the load rejection controller to the plant trip controller. The reactor trip signals are redundant.

During hot shutdown conditions, the steam dump system can be operated in the steam generator pressure control mode. Under these conditions the steam dump system is set to maintain a selected pressure in the steam generators.

When the transient results in a unit trip, the operator transfers dump control to the pressure mode and the valves are regulated by a main steam pressure signal to maintain no-load pressure. Lower pressures can be maintained automatically by manual adjustment of the pressure setpoint by the operator in the control room. This remote manual operation is performed during startup, cooldown, and reactor physics test periods.

During primary plant cooldown, the steam dump system is operated in the steam generator pressure control mode. The pressure set point is manually reduced to achieve the required cooldown rate.

All dump valves fail closed on loss of control and they are prevented from opening on loss of condenser vacuum. During a loss of condenser vacuum, excess steam pressure is relieved to the atmosphere through the power relief valves or the safety valves, or both.

Valves are blocked or prevented from opening on low-low Tavg by redundant signals which are used to block valve opening. Three cooldown valves may be opened by the control board switches when there is a low-low Tavg, but the other nine dump valves stay blocked.

An adequate drainage system is provided upstream from each dump valve. The dump lines are normally stagnant and therefore produce condensate continuously. This condensate is automatically removed to permit proper system operation.

10.4.4.3 Safety Evaluation

The steam dump system is not essential to safe operation of the plant. It is required, however, to give the plant flexibility of operation.

Because the dump valves are subjected to modulating control, each of them is provided with isolation valves to permit maintenance. The flow capacity of each valve is selected to prevent excessive cooldown should the valve fail in the open position.

When all the valves are out of service, the steam generator safety valves provide the relieving capacity required to maintain the steam system within the design limits.

No effects of pipe breaks are considered since all piping is either located in the Turbine Building or in other areas where the effect of pipe breaks would not jeopardize the safe shutdown of the plant.

The steam dump system blowdown connections are located on the main condenser steam inlet sufficiently high above the condenser tubes in order to prevent direct impingement of the

horizontally discharged high velocity steam on the condenser tubes or other condenser components; thereby precluding their failures from the steam dump system blowdown (See Figure 10.4-21).

Failure of the turbine bypass (steam dump) valves to open on demand does not affect operation of the turbine-generator unit, except that such a failure may cause a reactor trip (and subsequent turbine-generator trip) during primary to secondary plant power mismatches.

Failure of the turbine bypass (steam dump) valves to close will also lead to a turbine-generator trip following the reactor trip generated by low steam line pressure.

10.4.4.4 Tests and Inspections

Each dump valve is periodically tested. The test can be performed during power operation by shutting the associated isolation valve and locally checking for performance and timing. The dump valves are also operated during initial startup and during shutdown.

The steam dump lines are hydrostatically tested to confirm leaktightness.

Exterior visual inspection is used to check the condition of welds.

The steam dump valves were required to meet the following requirements during initial shop testing:

1. Hydrostatic tests in accordance with MSS SP-61

2. Seat leakage tests; 0.01 percent of maximum valve capacity with water at shutoff pressure

3. Operator leak tests; no visible leakage

4. Hysteresis and stroke speed tests at 25, 50, and 75 percent for upward and downward travel

10.4.4.5 Instrumentation Requirements

Indicating lights are provided in the Control Room for each dump valve to indicate when the valve is fully open or fully closed. They are also provided with local position indicators.

The magnitude of the modulating dump valve signal is indicated on the control board.

Control board lights are provided to indicate the status of permissive and interlock circuits.

Detailed descriptions of the steam dump instrumentation and controls are provided in Section 7.7.1.8. The logic diagram is shown on Sheet 10, Figure 7.2-1.

The low-low average temperature interlock (P-12) for steam dump block is the only safety-related instrumentation in this system and is designed in accordance with IEEE 279-1971.

10.4.5 CIRCULATING WATER SYSTEM

10.4.5.1 Design Bases

The Circulating Water System supplies approximately 1,100,000 gpm of cooling water to each unit. This flow is sufficient to remove the heat from the main condenser, the two auxiliary condensers, the turbine plant cooling water heat exchanger, the three condenser exhausting vacuum pump heat exchangers, and two Unit 1 or three Unit 2 non-safety ventilation chillers. The total heat removed amounts to approximately 8.8×10^9 Btu/hr, of which about 8.4×10^9 Btu/hr is removed from the main condenser. The Circulating Water System is supplied by the Squaw Creek Reservoir (SCR), which provides water at a design temperature of 95°F. The expected discharge temperature is an approximately 15°F temperature rise above the inlet temperature of SCR. The system is designed to operate with the water in the SCR at its lowest elevation of 770 feet.

10.4.5.2 System Description

Location of the circulating water discharge structure is shown on Figure 1.2-1. The Circulating Water System itself is shown on Figure 10.4-5. Water from the SCR flows to the eight circulating water pumps (both units) through heavy, steel bar trash racks and 12 traveling screens. The trash racks remove any heavy debris from the intake water while the traveling screens remove smaller debris which can also be present. The screens receive periodic cleaning, which is initiated automatically by means of a timer. When the screens become so clogged as to require cleaning prior to or during the periodic cleaning, a high differential pressure is indicated and backwashing is automatically initiated. If the differential pressure increases beyond the set point for automatic backwashing, an alarm is annunciated in the Control Room.

The backwash water is filtered and returned to the reservoir. For maintenance purposes, each screen well is provided with stop logs to allow dewatering of any individual screen well.

For each unit, the water from six screen wells flows to a common suction pit. Four motor-driven, vertical, centrifugal, mixed flow, circulating water pumps take suction from this pit. Each pump has 25 percent of the maximum capacity required for operation of one unit. This system is duplicated for operation of the second unit.

Both inlet and outlet connecting lines to the water boxes of the main and auxiliary condensers and the turbine plant cooling water heat exchanger are equipped with butterfly valves and expansion joints. Respective to each unit, one line branches off the intake tunnel and supplies water, which services the auxiliary condensers, the turbine plant cooling water heat exchanger, the condenser exhausting vacuum pump heat exchangers, and the non-safety ventilation chillers. There is a bleed line, which branches off the intake manifold of each unit, that can supply makeup water to the Safe Shutdown Impoundment (SSI).

The anticipated water quality of SCR is given in Table 10.4-3. Based on the anticipated water quality of the SCR, calculation of both stability and saturation indices indicate that the water tends to form scale and is corrosive.

Materials selected for use in this system are those that best withstand any long-term corrosion, erosion, pitting, or stress cracking when used with this water. The circulating water system is

composed of stainless steel, plastic, and carbon steel piping. Piping which is stainless steel, plastic, or carbon steel less than or equal to 2" in diameter is unlined. Carbon steel piping 2½" in diameter and larger is epoxy-lined. The main condenser water boxes are epoxy-lined. Titanium tubes are used throughout the condenser tube bundle assemblies in both the air removal and condensing sections. In addition, titanium clad carbon steel is utilized on the water box side of all tube sheets to minimize galvanic attack.

10.4.5.3 Safety Evaluation

The circulating water pumps are designed to remain operable with the water level at its lowest anticipated elevation. The pump motors and valve motor operators are supported so that no electrical parts are immersed in water at the highest anticipated elevation of the reservoir. See Section 2.4.11.6.

The Circulating Water System is not required for emergency cooldown or for operation of the engineered safeguard systems; instead, the Station Service Water System and the related SSI fulfill these functions.

The Circulating Water System is not required for cooling during shutdown because Reactor Coolant System (RCS) cooldown can be accomplished by dumping steam to the atmosphere through the power-operated main steam relief valves.

The results of a failure and its effects on safety-related systems will not adversely affect plant safety under any operating condition. Failure of the main condenser circulating water expansion joint can cause rapid flooding of the Turbine Building sump. Consequent flooding of safety-related equipment is prevented by tripping the Circulation Water Pumps before the flooding overflows the Turbine Building pit (elevation 778 feet, see Figures 1.2-26 and 27). A two out of three level trip logic will be used to automatically trip the Circulating Water Pumps for the affected unit.

The level trip has the following features to assure reliability:

1. The power supply for the trip circuit is an uninterruptible power supply 10kva inverter IV1C2 or inverter IV2C2 (Figure 8.3-13).

2. An alarm in the control room will be caused by a high water level trip condition or test bypass switch in the bypass position.

3. The level switches are the same as those qualified for 1E service. The qualification data is listed in CPNPP/EQR Table 5-1 (Purchase Spec MS-620).

4. The cabling from the level switches is in conduit and routed to protect it from jet impingement.

On shutdown of a circulating water pump, water hammer is avoided by ensuring that the pumps coast down as the pump isolation valves close. These valves are 90-inch-diameter butterfly valves which close in approximately 90 seconds. Condenser inlet and outlet water box isolation valves are also provided. On simultaneous trip of all four circulating water pumps, with no power available for closing the pump isolation valves, water hammer is avoided by a vacuum breaker on the outlet condenser water boxes. On startup, water hammer is avoided by the vacuum

priming of the Circulating Water System. This insures that air which can cause large pressure transients is removed. The valve is then opened, and the pump is started when the valve reaches 35 degrees open. The condenser tubes and water boxes are designed for a pressure of 40 psig, which is well above the maximum discharge pressure of the circulating water pumps.

The environmental considerations associated with disposal of waste heat from the CPNPP require dispersion of the warm condenser water in the SCR in order to provide protection of the aquatic life in the reservoir waters.

For each unit, the circulating water is returned to the SCR via a tunnel discharging into an open structure. This discharge structure greatly reduces the velocity of the circulating water from the pipeline to the end of the structure where the water flows into the reservoir. The discharge velocity is approximately 9.8 ft/sec. The low discharge velocity encourages stratification of the heated circulating water. This in turn promotes dissipation of the rejected heat by evaporation and heat transfer to the atmosphere. This mechanism involves the minimum amount of reservoir water in the heat dissipation process.

Corrosion effects are expected to be minimal because of the anticipated water quality and the material selection.

The circulating water is shock treated with a solutions of sodium hypochlorite and sodium bromide to reduce organic fouling.

10.4.5.4 Inspection and Testing Requirements

All active components of the system are accessible for inspection during station operation.

Model tests of performance were conducted in accordance with the Hydraulic Institute standards.

10.4.5.5 Instrumentation Requirements

A high differential pressure alarm is provided in the Control Room, across each condenser waterbox, to inform the operators of a water flow restriction. Local pressure, level, and temperature indicators at the waterbox inlet and outlet are provided. Plant process computer inputs are provided in the Control Room to record inlet and outlet water temperatures.

The circulating water pumps are individually equipped with motor-operated isolation valves. The main and auxiliary condenser waterboxes and TPCW heat exchangers are equipped with gear-operated isolation valves which enable the equipment to be isolated.

The circulating water pump discharge valves are interlocked to open and close on pump start/stop signals.

There is no safety-related instrumentation in this section.

10.4.5.6 Circulating Water Chemical Treatment Subsystem

10.4.5.6.1 System Description

Circulating water at CPNPP is drawn from the SCR and returned without concentrating dissolved solids or adding chemicals for pH and corrosion control. A solution of sodium hypochlorite and sodium bromide is added to the water to control organic and biological growth in the cooling system.

Figure 10.4-6 shows the major components and flow path for the circulating water chemical treatment subsystem. Sodium hypochlorite and sodium bromide are drawn from storage tanks by 100% capacity pumps and distributed to the circulating water at the intake bays.

10.4.6 CONDENSATE CLEANUP SYSTEM

The Condensate Cleanup System (CCS) of the CPNPP is provided for the removal of both dissolved and suspended solid contaminants in the condensate. The water used for backwashing may be discharged or reclaimed. The system components are located in the Turbine Building at elevations 803 ft and 778 ft, except for the hot phase separator, which is located in the Fuel Building at elevation 860 ft. One CCS is provided for each unit, with the exception of the hot phase separator which serves both units.

This system is not nuclear-safety-related. However, the CCS can, at times, be exposed to low levels of radioactivity caused by leakage of the primary coolant (in the steam generators) to the secondary coolant (condensate). Many radionuclides behave as dissolved solids and thus are adsorbed by the ion exchange resin. The system components are therefore designed to be resistant to the effects of short term exposure to low radioactivity levels. Seismic design criteria are not considered.

10.4.6.1 Design Bases

1. Flow Rate

Total system	100 percent of the condensate flow equal to 21,000 gpm at 150°F
Each vessel	5250 gpm (with four vessels operating in parallel and one vessel on standby)

2. Impurity Levels

The effluent water quality meets the secondary feedwater chemistry specifications of the EPRI guidelines, except where CPNPP evaluations justify exceptions.

3. Codes

The design codes applied to the CCS are as follows:

General	OSHA

Pressure vessels	ASME B&PV Code, Section VIII
Atmospheric tanks	API 620
Piping	ANSI B31.1
Materials and testing	Applicable ASTM and ASME standards
Electrical and control	Applicable IEEE and ISA standards

10.4.6.2 System Description

1. General

The CCS process flow diagram and equipment layout is shown on Figures 10.4-7 and 1.2-23 respectively.

The CCS normally uses powdered ion exchange resin coated on special filter elements to simultaneously filter and demineralize the condensate.

The main components of the CCS are polisher vessel skids, precoat tank and pump skids, overlay tank and pump skid, backwash pumps and tanks, spent resin tank and pump skid, conductivity sampling equipment, and a Programmable logic controller (PLC) panel and instrumentation.

The system has the following four subsystems:

a. Main cleanup vessels

b. Backwash and precoating facilities

c. Backwash recovery/waste facilities

d. Resin disposal facilities

2. Normal Operations

The condensate flows through the inlet header to the online demineralizer vessels. A flow metering and balancing system regulates the condensate flow through each vessel. The condensate enters each filter vessel in service from the outside to the inside of several hundred filter elements. The powdered resin coating is located on the outer surface of the filter elements. The processed condensate then flows to the system outlet header.

As suspended and dissolved matter are picked up by the resin, the pressure drop across the vessel increases. At a preset pressure drop, a high differential pressure alarm sounds notifying the operator to remove the vessel from service.

The conductivity and other selected chemistry parameters of each vessel outlet is monitored. A vessel high-conductivity alarm is available to notify the operator to remove

the vessel from service. Normally the vessels are changed out based on steam generator chemistry, vessel outlet conductivity and sodium, vessel differential pressure, and/or run time. Also, the sodium ion concentration of a single selectable vessel outlet is monitored.

During low flow or off-stream (standby) conditions, a holding pump recirculates condensate at a nominal flow required to maintain the resin coating on the elements of that vessel. The pump is automatically controlled by a low flow alarm signal.

3. Backwash and Precoating

An automatic system is provided to conduct the backwash and precoat operations of the off-stream vessel in the following sequence: the vessel is backwashed, vented, rewashed, filled, precoated, and placed in a standby condition or put into service. The backwash pump takes suction from the demineralized water storage tank and discharges through the vessel to the vessel backwash drain line.

The backwash air blower supplies air under pressure to the backwash header. The water air mixture flows from the inside to the outside of the filter elements, scouring the resin coating from the elements' outer surface. The backwash effluent is directed to the drain and finally to the A tank of the recovery system.

During this process, the vent valve remains open, allowing air to escape from the vessel. The vessel vent is also directed to the A tank. The backwashing process is repeated again; this is known as the rewash step. The rewash effluent is directed to the B tanks. The rewash step, which may be repeated, is of slightly longer duration than the backwash step. The backwash water from the cycle is normally discharged. The pump and the air blower are then stopped, and the vessel is drained. Then refilled with demineralized water.

The next stage of operation in the automatic cycle is precoating. The precoat media slurry, normally powdered resin, is pumped by the precoat pump through the precoat inlet header to the vessel elements where the resin is deposited on the elements and following the precoat cycle, is kept in position by the flow from the holding pump. The precoat water is routed back to the precoat mixing tank and the automatic cycle is terminated by putting the regenerated vessel in a standby condition.

4. Backwash Recovery

The backwash recovery system, working in conjunction with the exhausted vessel, may be used in incidents of severe water shortage to recover approximately 95 percent of the backwash water and produces a settled spent resin sludge of about 20 percent by weight.

Prior to backwash and precoating of the exhausted vessel, the backwash recovery process is initiated. The operator initiates the sequence to open the appropriate valves and to start the backwash recovery pump which draws water from the A tank supernatant and both B tanks.

Recirculation is established from the backwash recovery pump through the exhausted Powdex vessel back to the A or B tank. This recirculation serves to check the recovered backwash water conductivity.

Subject to an acceptable recovered water quality, the flow is diverted to the condenser deaeration section until the contents of the backwash recovery A or B tanks have been depleted, as signaled by closure of a low-level switch contact.

In the event of an unacceptable conductivity test, or other applicable chemistry parameters, for the recovered backwash water, a small quantity of powdered resin is mixed in the overlay tank, to be applied to the exhausted precoat in the vessel, for the purpose of increasing the ion exchange capacity of the exhausted vessel. The overlay step terminates upon closure of the low-level switch on the overlay tank, and the regular recovery process proceeds. If the total suspended solids count is higher than parameters allow then a temporary filtration system may be used to filter the total suspended solids out. This situation would occur in the event of a primary to secondary reactor coolant leak.

5. Resin Disposal

The A tank is used to hold the resin laden portion of the backwash stream and other streams liable to contain resin. The A tank receives 95 percent or more of the spent resin precoat. The conical bottom and the variable speed scraper of the A tank helps settling and thickening of the resin slurry. The settled resin slurry is then pumped by the spent Powdex pump to the phase separator tank. The two B tanks are used for holding the resin free portion of the backwash until it is discharged or recovered.

The phase separator tank is used as a final settling and decanting tank of the A tank resin slurry. The phase separator resin slurry underflow is pumped by the phase separator pump to the Low Volume Waste Treatment Facilities while the liquid portion decants or overflows to the A tank. Requirements for sampling the resins for radioactivity prior to transfer to the LVW Treatment Facilities and a limit on the total inventory of radioactive material contained in powdex resins allowed in the pond are established in the Radiological Effluent Controls Program required by the Technical Specifications.

When activity in the resin slurry exceeds the Radiological Effluent Controls limits, the resins are transferred to a disposal container for processing as radioactive solid waste.

10.4.6.3 Equipment Design

The design criteria for the Holding Pumps and the Polisher Vessels is a design pressure of 600, psig. The design pressure for the Backwash Pump, Precoat Pump, Backwash Recovery Pump, Spent Powdex Pump, and Phase Separator Pump is 150 psig. The Precoat Tank, Overlay Tank, Backwash Recovery A Tank, Backwash Recovery B Tank, Phase Separator and Hot Phase Separator are designed for Atmospheric Pressure. The Safety Class for all the previously referenced equipment is NNS.

10.4.6.4 Safety Evaluation

Since the CCS is not required for maintenance of plant safety, the system can be bypassed for short periods of time (provided there is no condenser leak, or chemistry upset) without significantly affecting plant operations. The system permits limited plant operation during condenser inleakage (depending on the magnitude of the leak) by reducing contaminants in the condensate to minimize the impact on steam generator chemistry.

The effluent water quality meets the secondary water chemistry specifications of the EPRI secondary water chemistry guidelines, except where CPNPP evaluations justify exceptions.

It is estimated that the contribution of impurities from the secondary side of the RCS is negligible because of the use of the CCS, SGBS, all volatile secondary-side chemistry specifications, and infrequent occurrence of pressure differentials for secondary to primary leakage.

Continuous conductivity and differential pressure monitoring ensure the proper functioning of the CCS.

10.4.6.5 Instrumentation and Control

1. Flow Instrumentation

   A flow control system is provided to equalize the flow rate of condensate to each onstream vessel. Vessel flow rate is regulated by a proportional signal from the flow control system to the vessel outlet valves. To continually maintain the precoat on the filter elements, holding pumps (one for each vessel) are activated when flow to a vessel decreases below a preset minimum. Flow instrumentation also includes rate-of-flow indicators for backwash water and air.

2. Pressure Instrumentation

   Differential pressure across each vessel and the entire system is continuously monitored. High differential pressure across a filter vessel causes an alarm to sound, indicating to the operator that the service run of that vessel should be terminated. By using a differential-pressure-indicating controller to automatically open the system bypass valve on increasing system differential pressure, a pressure drop of $\leq$40 psi is constantly maintained.

3. Conductivity Instrumentation

   Cation conductivity is monitored at the system inlet, system outlet, and at each vessel effluent outlet. High conductivity at any analysis point sounds an alarm to notify the operator to either remove a vessel from service or to look elsewhere in the condensate system for possible trouble (i.e., condenser leakage).

   The conductivity rack also contains a stainless steel sample sink for manual sampling.

4. Control Panel

Two redundant state-of-the art local Programmable Logic Controller (PLC) panels, each with an Operator Interface (CRT based) Unit, are provided for each CCS. The two PLCs are cross linked to serve as a back-up for each other. They are mounted independently of the system skids.

The PLCs operate all CCS equipment automatically and precisely, cycle after cycle. Precoat and Backwash cycles are controlled automatically. Display of equipment status, indications of process variables, and alarm status is provided on the CRTs. They also provide Data Acquisition, Trending and Report Generation.

A common trouble alarm is provided in the Control Room.

5. Sodium Instrumentation

Sodium ion concentration is monitored locally at each vessel effluent. A single sodium ion analyzer is provided for this purpose. Each individual vessel effluent may be selected for sodium ion analysis.

10.4.6.6 Tests and Inspections

The polisher vessels and piping are hydrostatically tested in the assembly shop at 1.5 times the system design pressure. The control system is also tested in the shop to ensure proper valve and pump operation sequences. After installation, the system is again hydrostatically tested.

The filter elements are inspected both when installed and at periodic intervals for physical defects which could cause excessive pressure drops or loss of powdered resin. The expected life of the filter element is five to eight years.

In the event the backwash recovery is used, conductivity, millipore, and/or other applicable tests are conducted on the recovered backwash water to determine whether it should be diverted to the condenser deaeration section or be subjected to further processing.

10.4.6.7 Other Features

The CCS design takes into consideration portions of the recommendations of NRC Regulatory Guide 1.56, positions 1 through 4, to the extent of their applicability to PWRs with recirculating-type steam generators. Position 5 is not applicable for this system.

The CCS is located in the Turbine Building. Figure 1.2-23 of the CCS and the Turbine Building shows that no nuclear-safety-related component or system is located close to the high-or moderate-energy pipes of the CCS. The design of the CCS meets Branch Technical Positions APCSB 3-1 and MEB 3-1, as detailed in Section 3.6.

10.4.7 CONDENSATE AND FEEDWATER SYSTEMS

The Condensate and Feedwater Systems return the condensate from the turbine condenser hot wells through the regenerative feed heating cycle to the steam generators while maintaining the water inventories throughout the cycle.

10.4.7.1 Design Bases

The Condensate and Feedwater Systems are designed to provide approximately 16.32 x 10^6 lb/hr of feedwater at 446.7°F to the steam generators during steady-state operation at the 100% NSSS power (3628 MWt) with 66°F Circulating Water Inlet Temperature.

The condensate portion of the system is designed to supply approximately 9.976 x 10^6 lb/hr to the suction side of the steam generator feedwater pumps during steady-state operation at maximum guaranteed turbine load. In addition, the Condensate System in conjuction with the Heater Drain System can supply 96 percent of the full-load feedwater flow (15.667 x 10^6 lb/hr) to the steam generator feedwater pumps during load-drop transients.

The feedwater portion of the system is designed to supply the feedwater required for various loads at steady-state operation and to maintain this flow, as required, during the steam dump conditions following a large load reduction. The system is designed to maintain uniform feedwater flow to all steam generators under all conditions and to maintain proper steam generator water levels automatically during steady-state and transient conditions.

The Condensate Storage Tank is designed to store supply water for the Condensate and Feedwater Systems and the Auxiliary Feedwater System (see Section 9.2.6).

Condensate is transferred from the auxiliary condensers to the main condenser shell A hot well by gravity. The design parameters for the major components of the Condensate and Feedwater Systems are given in Table 10.4-5.

The only-safety related portion of the Condensate and Feedwater Systems is the feedwater piping portion between the moment restraint upstream of the Feedwater isolation valve and the steam generator feedwater nozzle. This portion is designed to seismic Category I requirements and is designed to withstand the worst anticipated environmental phenomena taken individually. The Condensate Storage Tank is designed to seismic Category I requirements and designated Safety Class 3. It is considered part of the Auxiliary Feedwater System.

The Condensate and Feedwater System components are designed and constructed in accordance with the following applicable regulations, codes, and standards:

1. Code of Federal Regulations, 10 CFR Part 50

2. Branch Technical Positions APCSB 3-1 and MEB 3-1 [6] [7]

3. ASME B&PV Code

Section III	Rules for Construction of Nuclear Power Plant Components, Division I, subsection NC, Class 2 Components
Section XI	Rules for Inservice Inspection of Nuclear Power Plant Components
Section VIII	Pressure Vessels (Division 1)

4. NRC Regulatory Guides

1.26	Quality Group Classification and Standards for Water, Steam & Radioactive-Waste containing components of Nuclear Power Plants. [3]
1.29	Seismic Design Classification [5]
1.32	"Criteria for Safety Related Electric Power Systems for Nuclear Power Plants" [22]
1.47	"Bypassed & Inoperable Status Indication for Nuclear Power Plant Safety Systems" [23]
1.53	"Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems" [24]
1.75	"Physical Independence of Electric Systems" [25]

5. American National Standards Institute (ANSI)

B31.1	Power Piping
N18.2	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants [4]
N271	Containment Isolation Provisions for Fluid Systems

Additional requirements for the portion of the Feedwater System which is within the Containment boundary are given in Section 10.3.6.

10.4.7.2 Systems Description

The Condensate and Feedwater System flow diagrams are shown on Figure 10.4-8, and 10.4-9, respectively.

1. Condensate System

Two motor-driven, constant-speed, vertical, canned-type condensate pumps are supplied, each designed for approximately 65 percent of the total condensate flow. These pumps withdraw condensate from the two main condenser hot wells via a common discharge arrangement which cross connects the two condenser shells, minimizing level differences between the two hot wells.

Both pumps are vented to main condenser shell B to prevent air binding in the pumps. Seal and priming water are supplied to the condensate pumps from the Condensate Storage Tank. The condensate pumps discharge into a common 30-in. header, which carries the flow to the Condensate Polishing System full-flow condensate filter-demineralizer vessels. The Condensate Polishing System is described in Section 10.4.6.

The total condensate flow rate is measured in the 30-in. line from the condensate pump discharge header to the condensate filter demineralizers. This measurement is made to ensure that the flow rate does not fall below the minimum flow requirements of the condensate pumps (2900 gpm per pump). Should the demands of the Feedwater System be low, the recirculation line valve upstream from the drain coolers opens, allowing the additional flow to return to the condenser.

At the outlet of the filter demineralizers, the flow divides to provide cooling water to both the main and auxiliary gland steam condensers under normal- and minimum-recirculation condensate flow conditions. These lines rejoin the balance of the condensate flow in the 30-in. header downstream of the main and auxiliary gland steam condensers. The flow is then divided into two approximately equal streams, each passing in succession through the tube side of the drain cooler and two stages of regenerative heating provided by feedwater heaters Nos. 5 and 6. A cross connection between the inlets of heaters 4A and 4B promotes equalization of temperature. The flow remains separated as it passes through the next two stages of feedwater heating provided by feedwater heaters Nos. 3 and 4.

The condensate flow is then distributed equally to the suction side of the feedwater pumps after passing through a flow measuring device, located in each pump suction line, used to control feedwater pump recirculation. Specific pairs of feed water heaters in each train are provided with normally closed bypass valves which allow for the removal of these heaters from service for maintenance or repair.

An emergency low-pressure heater bypass is included in a line branching from the condensate pump discharge header upstream from the Condensate Polishing System. The line discharges downstream of the flow element located in the heater drain pump discharge line. The flow is then directed to the steam generator feedwater pump suction lines. Under transient conditions, such as a 40-percent steam dump, fluid in the heater drain tanks may flash, causing a low water level and tripping the heater drain pumps resulting in the loss of their feed flow to the feedwater pump suctions. The loss of this flow (6,346,011 lb/hr at full load) would rapidly lead to a loss of pressure at the feedwater pump suctions. This in turn would result in the loss of feedwater flow, and then a turbine trip. To avoid a turbine trip, the bypass is sized so that 15,667,000 lb/hr (96 percent of full flow) is available to the feedwater pumps for supplying steam generator requirements during a load drop transient. The bypass valve opens fully within 2.5 seconds of receipt of the initiating signal.

The exhaust steam from the steam generator feedwater pump turbine drivers is condensed in the two small auxiliary condensers. The condensate from the auxiliary condenser hot wells normally drains under gravity to the main condenser hot well.

Condensate is obtained from the Condensate Storage Tank upon receipt of a low-level signal from the main condenser shell B hot well. Excess condensate is returned to the Condensate Storage Tank through the condensate pump reject control station.

The 150-gpm auxiliary boiler feedwater pumps obtain suction from the condensate makeup and rejection line.

Demineralized water from the Demineralized and Reactor Makeup Water System, which serves both units of CPNPP, can be introduced into the main condenser hot wells via the Condensate Storage Tank.

All normally open Condensate System Condensate Storage Tank connections are set 23 ft 9 in. above the bottom of the tank to prevent automatic drawdown below this level, thus ensuring that a storage of approximately 282,000 gal of water remains in the tank for possible emergency plant shutdown which exceeds the Technical Specification volume requirements. This includes 12,900 gallons of residual capacity which is below the critical depth and is therefore not considered available.

The design basis for the minimum storage capacity is as follows:

a. a volume of water required to maintain the reactor at hot standby for 2 hours followed by a cooldown to 350°F at a rate of 50°F/h for 5 hours with allowance for feedwater line break spillage.

b. volume of water required to maintain the reactor at hot standby for 4 hours followed by a cooldown to 350°F at a rate of 50°F/h for 5 hours.

All heaters are provided with tube side-safety valves to provide thermal relief by guarding against possible overpressurization caused by heating of water trapped between closed isolation valves.

2. Feedwater System

The Feedwater System is of the closed-cycle type and receives water from the Condensate System and the Heater Drains Systems (specifically, drains from heaters 1, 2, and 3, and MSR Separator drain tanks). The feedwater is transported through the final two stages of feedwater heating to the steam generators.

During power operation each steam generator feedwater pump takes suction from the Condensate System and discharges through a common header to the high-pressure feedwater heaters. One steam generator feedwater pump is required to operate during power operation of up to 50% of the rated power. From 50-percent power to full power, both steam generator feedwater pumps are required, with each pump providing 50-percent of the required flow. Prior to aligning feedwater pumps to the steam generators, water is supplied by the Auxiliary Feedwater System.

Each steam generator feedwater pump is fitted with a minimum recirculation control system which protects the pumps from damage at low loads by ensuring a minimum flow.

Leakages through the pump are detected by monitoring the temperature of the seal injection system drains.

The dual admission feedwater pump turbine drivers operate with steam from two sources. During low-load conditions, high pressure steam is supplied to the turbines from the Main Steam Supply System steam dump header. During normal operation, low-pressure steam is supplied from the MSR in the Main Steam Supply System. Gland steam is provided to the turbines from the turbine gland steam seal supply system.

The flow from the two steam generator feedwater pumps combines at the pump discharge, then divides into two streams for the final two stages of regenerative feedwater heating (heaters Nos. 1 and 2). The two heater trains and the common bypass join downstream of the high-pressure heaters to form a single common header for temperature equalization. From this common header, an individual feedwater line supplies each steam generator.

When originally constructed both Unit 1 and Unit 2 incorporated a Feedwater Bypass System on each main feedwater line to a steam generator. The function of the Feedwater Bypass System is to minimize the potential occurrence of a water hammer in the steam generators, and to mitigate the flow induced tube vibration in the steam generators.

With the installation of Delta 76 steam generators on Unit 1, the Feedwater Bypass System is no longer required for the new feedring steam generators and has been removed from Unit 1 with the exception of the small bypass line around the main feedwater isolation valves. (Note: The Unit 1 feedwater system is operated in a different manner after RSG installation. The FIV remain open during startup and a small flow through the valve and not the FIBV is used for purging purposes.) Unit 2 still retains the Feedwater Bypass System in its entirety.

The Unit 2 Feedwater Bypass System associated with an individual Main Feedwater line consists of three lines with associated instrumentation and controls as follows:

The Unit 2 feedwater preheater bypass line connects each main feedwater line, just upstream of the main isolation valve, to the auxiliary feedwater nozzle in the upper portion (above normal water level) of each steam generator. Each bypass line has its own containment isolation valve (Feedwater Preheater Bypass Valve) which is air operated. The objective of this bypass line is to minimize the potential occurrence of pressure transients and to prevent feedwater from entering the main feedwater nozzle at startup and during certain other operating conditions.

The Unit 2 feedwater split flow bypass line connects the main feed line to the feedwater preheater bypass line inside containment to provide a continuous feedwater split flow during normal plant operating conditions in order to minimize thermal transients in the nozzle and connecting piping when flow is transferred to the auxiliary FW nozzle from the main FW line and to minimize the flow induced tube vibration in the steam generators. It contains an Annubar flow element and an air operated butterfly valve.

In addition, a flow restricting orifice has been installed in each Unit 2 main feedwater line just downstream of the bypass line connection to facilitate the required flow split.

In addition, a small bypass line around the feedwater isolation valve is provided to purge cold feedwater from the main feedwater line between the isolation valve and the steam generator feedwater nozzle. This line incorporates a restricting flow orifice and an air operated globe type shutoff valve which also serves as a containment isolation valve.

Connections from the Condensate System are provided both upstream and downstream of the heaters to permit flushing of the heaters. The upstream connection can also be used to fill the system and the steam generators and to provide feedwater directly from the condensate pumps during the early stages of startup.

Full flow flushing of the Condensate and Feedwater Systems is provided to permit piping cleanup before plant start-up. Full flow flushing is accomplished through a 24" diameter pipe routed from the outlet of the HP feedwater heaters to the condenser. Bypass lines with isolation valves are provided around each steam generator feedwater pump for use during flushing operations.

Condensate and feedwater chemistry are controlled as described in Section 10.3.5. This system uses all volatile chemical treatment in conjunction with a Condensate Polishing System, which is described in Section 10.4.6.

Chemical feed to Unit 2 steam generators is introduced at a point upstream of the auxiliary feedwater supply connection to the Feedwater System for adjustments to the water chemistry during steam generator layup periods only.

3. Electrical Systems

Safe shutdown of the plant does not rely upon the availability of either the Condensate System or the Feedwater System. However, the portion of the Feedwater System from the moment restraint upstream from the feedwater isolation valve to the steam generator feedwater nozzle is nuclear-safety-related and automatic isolation of these lines is required during emergency conditions. The split flow bypass line (Unit 2 only) also requires isolation during emergency conditions. To ensure isolation of these lines, the feedwater isolation valves, feedwater isolation bypass valves, feedwater split flow bypass valves (Unit 2 only), and steam generator preheater bypass valves (Unit 2 only) are all tripped closed by separate, redundant control circuits and solenoids and are powered from two Class 1E-125 VDC power systems, maintaining the proper train separation. The independence and redundance of these two Class 1E-125 VDC power systems is described in Section 8.3.2.

In addition, the feedwater control valves and feedwater control bypass valves are also tripped closed by separate, redundant control circuits and solenoids and are powered from two class 1E 125 VDC power systems, maintaining the proper train separation.

The rest of the valves and equipment in the condensate and feedwater systems are powered from non-Class 1E systems.

10.4.7.3 Safety Evaluation

The requirements of 10 CFR Part 50, GDC 57, for Containment isolation are satisfied by one stop valve on each feedwater line outside the Containment (see Section 6.2.4.). With loss of flow in the normal direction, the check valve upstream of the stop valve closes and is held closed by back pressure. In addition, the stop valve can be closed and secured by remote operation.

The Feedwater System from the steam generators, back to and including the moment restraint upstream of the feedwater isolation valve, is designated as Safety Class 2 and is designed to the requirements of seismic Category I systems (see Section 3.2.1). For an analysis of the effects of a break in the Safety Class portion of the Feedwater System, see Section 10.4.9. The remaining portion of the Feedwater System piping inside the Safeguards Building is designated non-nuclear safety seismic Category II. The portion of the system in the turbine building and the Condensate

System is non-seismic except for the Condensate Storage Tank which is designated Safety Class 3, seismic Category I.

The Condensate and Feedwater Systems are designed to the requirements of the codes listed in Subsection 10.4.7.1. The potential for pipe rupture caused by internal pressure and temperature is as discussed in Section 3.6. Short lengths of pipe are installed above the turbine operating floor and could be fractured by a heavy object carried by the gantry crane or by a turbine-generated missile; but such an occurrence is extremely unlikely. A rupture of the condensate piping anywhere in the Turbine Building does not cause failure of safety-related equipment as a result of flooding, as no such equipment is installed therein.

In the Safeguards Building the feedwater pipes are, for most of their length, enclosed in separate, reinforced concrete ducts fitted with individual drains. Elsewhere, floor drains are designed to collect any discharge from the pipe, preventing damage to safety-related equipment resulting from flooding.

Any failure in the non-safety-class portion of the Condensate and Feedwater Systems has no effect on the safety of the reactor, which can be shutdown in an orderly manner. (See Section 3.6 for a further discussion of postulated pipe rupture.) A source of feedwater supply to the steam generators is required for decay heat removal from the reactor following a unit shutdown. In the event that the Condensate and Feedwater Systems are not available, the Auxiliary Feedwater System (see Section 10.4.9) provides the required emergency supply of feedwater.

Condensate available for emergency purposes is stored in the Condensate Storage Tank (see Section 9.2.6), which is a seismic Category I structure (see Section 3.2.1).

Although unlikely, a small amount of radioactivity may be present in the Condensate and Feedwater Systems in the event of a steam generator tube leak. Water from a pipe leak or break in the Condensate and Feedwater Systems is collected by the Equipment and Floor Drainage System. These drains are monitored for radioactive releases and are handled as described in Section 9.3.3.

The Unit 1 Delta 76 RSGs incorporate a feedring design in lieu of a preheater design and this feedring design feature prevents or mitigates the possibility of a steam generator water hammer event (CP1-RSG-05-184).

Significant flow instabilities due to steam void collapse (i.e., feedline waterhammer) are not expected to occur in the Unit 2 main feedwater system during normal operating transients due to the geometry of the system. The pre-requisities for such events have been shown to be (Refs. 18 and 19); uncoverage of the main feedwater inlet nozzle to the steam generator, thus allowing steam to enter the feedwater line, in conjunction with injection of cold feedwater. The design of the steam generator employed on Unit 2 has the main feedwater inlet nozzle located close to the tube sheet (see Figure 5.4-4). Accident analysis indicates that uncoverage of the nozzle is limited to a highly improbable sequence of events associated with some faulted condition transients. The results of a comprehensive test program have been used to demonstrate (Ref. 19) that the worst case waterhammer under such conditions will not result in a loss of primary circuit pressure boundary integrity under associated faulted conditions. Nevertheless, additional means of preventing the conditions under which waterhammer may occur have been implemented. Loop seals located at the feedwater inlet nozzles to the Unit 2

steam generators minimize the volume of feedwater piping that could self drain and fill with steam (see Figure 10.4-22). Furthermore, system design modifications have been implemented on Unit 2 which preclude the introduction of cold feedwater through the main feedwater inlet nozzle. These modifications, described in Section 10.4.9, consist of a feedwater bypass system to a separate auxiliary feedwater inlet nozzle (Figure 5.4-4).

Tests have shown (Ref 19) that pressure transients, due to steam void collapse, can occur in the steam generator and main feedwater piping if feedwater below 250°F is supplied through the main feedwater nozzle concurrent with low steam generator water level (at or below the level of the main feedwater nozzle in the preheater region) or low steam generator pressure. Although analysis of these test results has demonstrated (Ref 19) that the maximum pressure transients produce stresses below the allowable limits, a feedwater bypass system has been incorporated on Unit 2 to minimize the possibility of steam generator preheater and feed water piping pressure transients. The feedwater bypass system consists of a connection between the auxiliary feedwater nozzle and the main feedwater line upstream of the feedwater isolation valve on each steam generator. This bypass line contains a feedwater preheater bypass valve. The feedwater bypass system also include a feedwater isolation bypass valve.

To minimize pressure transient potential, it is necessary to prevent the introduction of cold water to the Unit 2 steam generator through the main feedwater nozzle at any time when significant void may be present. Therefore, total feedwater flow is not aligned to the normal feedwater nozzle during a startup until the feedwater line temperature is above a set limit. Conversely, feedwater flow is diverted to the auxiliary feedwater nozzle from the main feedwater nozzle during a shutdown.

The feedwater split flow bypass line inside the containment, which connects the main feedwater line and the feedwater bypass line, is designed to minimize the thermal transients in the steam generator nozzles and to preclude the flow induced tube vibration in the pre-heater section of the steam generator (see reference 21) by maintaining a feedwater flow split during feedwater injection through the main feedwater line. Main feedwater flow and feedwater split flow bypass flow are monitored during power operations to ensure an acceptable flow rate relationship exists between the two flow paths. A high flow alarm is available to identify when the normal main feedwater flow is exceeded. A vendor assessment of the potential for pre-heater region tube wear due to associated tube flow induced vibrations has been performed; and guidance is provided to establish a flow-time limit criterion in the event of a flow excursion. In addition, selected steam generator tubes in the preheater section have been expanded at two support plate locations to minimize vibration (reference 20). To achieve the required flow split, each split flow bypass line has been modified to minimize hydraulic resistance by incorporating an air-operated butterfly valve and an Annubar flow element to monitor and modulate the flow rate.

In addition, a flow restricting orifice has been installed in each main feedwater line just downstream of the bypass line connection to facilitate the required flow split.

10.4.7.4 Tests and Inspection

Each feedwater heater, drain cooler, pump, and valve receives a shop hydrostatic test. Prior to initial operation, the completed Condensate and Feedwater System is to receive a field hydrostatic test and inspection. These tests are performed according to the applicable codes listed in Subsection 10.4.7.1. Periodic tests and inspections of the system are to be performed in conjunction with scheduled maintenance outages.

Periodic in-plant tests are conducted to demonstrate the ability of the feedwater isolation valves to respond to a test close signal. The valves must close within the specified time. The valves are designed and constructed with provision for periodic inservice testing of partial valve stroke. Provisions are made in the trip mechanism for all solenoid-operated valves to be exercised without interrupting availability of the trip mechanism.

10.4.7.5 Instrumentation Requirements

Condenser hot well level is maintained by adding water on low level by opening makeup control valves from the Condensate Storage Tank and rejecting water to the Condensate Storage Tank on high condenser hot well level. The makeup control valves are modulated open by split-range signals from a level controller in the control loop with a level transmitter on condenser B hot well. The condensate reject valve is opened and closed by level switches on condenser shell A hot well. Tripping of both condensate pumps will close the condensate reject valve.

Makeup water can be manually added by a control board hand auto station. Water can be rejected manually by an open close switch on the control board.

A condensate gland steam condenser bypass valve is used to prevent excessive flow through the gland steam condenser in Unit 2. An orifice in the main condensate line is sized to provide design flow to the aux and gland steam condensers (Unit 1).

Condensate pumps are tripped by a condenser hot well low level. Failure of motor-operated discharge valves to open to a minimum position will prevent pump start. Minimum flow protection is provided by an on-off recirculation control valve. If one condensate pump trips with the plant above 50 percent load, the result is an automatic runback of the main turbine. However, automatic turbine runback initiated by a condensate pump trip is not sufficient to preclude the unit from tripping.

For Unit 1, the emergency low pressure heater bypass valve opens in less than two and a half seconds after receiving two out of three coincident feedwater pump low suction pressure signals. For Unit 2, the emergency low pressure heater bypass valve opens in less than two and a half seconds after receiving potential two out of three coincident feedwater pump low suction pressure signals. Possible water damage to the turbine caused by high water levels in heater Nos. 5A, 5B, 6A, and 6B, located in the condenser neck, is prevented by means of motor-operated isolation and condensate bypass valves.

Minimum flow protection for heater drain pumps is provided by a modulating recirculation control valve; this valve closes on pump trip.

The feedwater pump turbine control maintains proper pressure differentials across the feedwater control valve by using the input signals of the steam generator header pressure and the set point differential pressure obtained as a function of the sum of all four of the steam loop flows.

In case of loss of feedwater control to the feedwater pump turbine or excessive noise, turbine speed signal protection is provided by an automatic signal shift to a speed signal memory which retains the last functioning signal. Loss of signal is alarmed, and the feedwater pump turbine speed automatically transfers to the manual mode of control (also alarmed).

The feedwater pump turbine interlocks include turbine trip on safety injection signal, turbine trip on high-high steam generator level, and turbine trip on low feed pump suction pressure. Trip of both turbines starts the motor-driven auxiliary feedwater pumps. Trip of a single turbine causes a runback of the main turbine. The turbine trip pressure switches, which are used to start the auxiliary feedwater pumps, are redundant Class 1E switches. Feedwater control and bypass valves are tripped closed by the same signals that trip close the feedwater isolation valves. Redundant, Class 1E, “close” solenoids and associated circuitry are provided for each valve assuring the transmission and receipt of a “close” signal at each valve via at least one of two trains in spite of any single failure in any one train.

The feedwater control valves are regulated by the Westinghouse control system. The Westinghouse system is a three-element control which uses signals from the steam generator water level, the steam flow, and the feedwater flow. Below 25 percent load, the feedwater control bypass valve maintains the steam generator water level by using a control from the steam generator water level. Transfer from the feedwater control bypass valve to feedwater control valve is manually initiated.

Feedwater isolation valves are tripped closed upon receipt of the following safety-related instrumentation signals: a steam generator high-high level (two out of three high-high level signals from any steam generator), a safety injection signal, or a low average temperature with reactor trip. When the plant is operating these valves can be tested for partial closure of the valve. Feedwater isolation bypass valves, feedwater control valves, feedwater control bypass valves and feedwater preheater bypass valves (Unit 2 only) are all tripped closed by separate, redundant control circuits and solenoids.

Based on section 10.4.7.3 Unit 1 does not require any instrumentation to mitigate feedline water hammer.

The Unit 1 feedwater system is operated in a different manner after RSG installation. The FIV remains open during startup and a small flow through the valve and not the FIBV is used for purging purposes.

Feed pump discharge valves are opened by feed pump turbine start signals and are closed by feed pump turbine trip signals. The trip signal overrides the start signal.

Process sampling valves are automatically closed with a Containment isolation phase A, train B signal from the Westinghouse system. Sampling means are provided to monitor the quality of the feedwater. Temperature measurements, provided for each stage of heating, include the temperature into and out of each feedwater heater for the water side of the system and temperature out for the steam side. Steam side temperature measurements into each feedwater heater are provided for extraction lines that may contain dry steam.

Pressure measurements for wet steam lines will suffice to give the saturated temperature in the line. Steam-pressure measurements are provided at each feedwater heater.

In order to preclude water hammer in the steam and feedwater systems, the following interlocks and controls are provided. These water hammer interlocks are non-safety grade controls. However, where the non-safety grade water hammer interlocks interface directly with safety class equipment electrical isolation devices preclude any adverse impact on the safety class equipment caused by potential failure of the non-safety equipment.

1. FEEDWATER ISOLATION VALVE (FIV)

In order to minimize the potential for water hammer, interlocks are provided to prevent the FIVs from being opened (if they are closed) or, to close them if the FIVs are open, and route the feedwater through the auxiliary feedwater nozzle via the preheater bypass line. All of the listed interlocks must be present and the absence of a feedwater isolation signal to allow the FIVs to open.

a. Feedwater Flow

Feedwater flow must be above a low-flow set-point, as measured by a flow switch at the feedwater flow venturi meters. FS-2189, -2190, -2191, and -2192 are provided for loops 1, 2, 3 and 4 respectively. The set-point for these flow switches corresponds to approximately 12 to 15 percent of full feedwater flow.

Once the flow permissives have been cleared allowing the FIV to open, the FIV can remain open irrespective of flow, providing the FW temperature remains high (above the set point as described in item (b) below). Interlocks are provided for this condition.

b. Feedwater Temperature

The feedwater temperature must be above approximately 250°F (as measured by resistance temperature detectors on the main feedwater lines). In addition, the difference in temperature between the RTDs installed outside containment, downstream of the FIVs and RTDs mounted at a piping low point on the feedwater lines inside containment, near the main feedwater nozzle must be within about 10°F of each other. This arrangement of temperature sensors is used to preclude pocketing of cold water at the piping low point during startups and, also, to avoid the possibility of a single RTD open circuit failure causing a false temperature permissive signal to open the FIVs.

Once the temperature permissives have been cleared allowing the FIV to open, the FIV can remain open irrespective of temperature, providing the FW flow remains high (above the low flow set point as described in item (a) above). Interlocks are provided for this condition.

2. FEEDWATER PREHEATER BYPASS VALVE (FPBV)

The FPBV bypasses feedwater from the main feedwater line to the upper auxiliary nozzle. It also serves as a containment isolation valve and is redundantly interlocked with the feedwater isolation signals provided by Westinghouse.

The FPBV is automatically opened whenever the FIV is closed due to the absence of water hammer permissive signals as described above for the FIV. Likewise, it is automatically closed whenever the water hammer permissives are cleared and the FIV is opened. It is also automatically closed on Feedwater isolation actuation signal. The FPBVs are provided with three position close-auto-open control switches on the main board. The switch spring returns to auto; the valves fail closed on loss of control power or air.

3. FEEDWATER SPLIT FLOW BYPASS VALVE (FSBV)

The FSBV is provided to maintain a split bypass flow from the main feedwater line to the upper auxiliary feedwater nozzle. This bypass valve is open at about 30 percent RTP due to system hydraulics when feedwater is admitted to the steam generators through the main feedwater nozzle with the Feedwater Isolation Valve open, and is used to minimize flow induced tube vibration in the preheater section as well as thermal transients at the auxiliary feedwater nozzle.

The FSBV receives an open permissive signal when water hammer interlocks are satisfied. Likewise, the FSBV is automatically closed when the water hammer interlock permissive signal is removed. The FSBV is also automatically closed on an AFW initiation signal to direct AFW to the auxiliary feedwater nozzle, to minimize delays in admitting AFW to the steam generator, and to prevent steam generator blowdown through the auxiliary nozzle during a main feedwater line break.

An annubar flow meter, downstream of the FSBV, measures the flow and provides a signal to a controller to modulate the valve. A control board flow indicator and a manual/auto station is provided for each FSBV. A high-flow alarm is provided to alert the operator when the maximum allowable flow into the preheater is exceeded.

4. FEEDWATER ISOLATION BYPASS VALVE (FIBV)

The FIBV is a containment isolation valve, of the air-operated, fail-closed type. It is redundantly isolated by the Westinghouse feedwater isolation signals from Train A and Train B.

The FIBV stays closed whenever the lack of water hammer permissive signals causes the FIV to close, and is automatically closed after the water hammer permissives are cleared and the FIV has been fully opened manually. The FIBV is manually opened to provide a means of purging the main feedwater line whenever the FIV is closed in order to preclude pockets of cold water from collecting in the main feedwater piping downstream of the FIV. The magnitude of the purge flow is limited by a restriction orifice upstream of the FIBV.

The FIBVs are controlled individually by main control board mounted, three-position, close-auto-open switches which spring-return to auto.

10.4.8 STEAM GENERATOR BLOWDOWN SYSTEM

Under all normal operational conditions, the Steam Generator Blowdown System (SGBS) is used in conjunction with the Chemical Feed and Sampling Systems to maintain optimum secondary-side water chemistry in the recirculating steam generators and to control radioactivity levels associated with nonvolatile radionuclides.

The system components are located in the Auxiliary Building at elevations 778 ft, 790 ft 6 in, and 831 ft 6 in. Each unit has a separate and independent SGBS.

10.4.8.1 Design Bases

10.4.8.1.1 General

Each SGBS is designed to continuously treat 100 percent of the blowdown from each unit. The treated blowdown is returned to the condenser or heater drain tank for reuse as secondary coolant. Typically the blowdown system is operated with the demineralizers in service.

The design primary-to-secondary system leakage is assumed to be 20 gpd, measured at 25°C. The reactor coolant activity is based on 1.00 percent fuel defect. Radioactivity is reduced by the demineralizers until the resins are exhausted. The resin bed capacities are determined by resin saturation with the primary plant chemicals, which greatly exceed the concentration of radioactive dissolved solids. Thus, the resin usage factor depends on neither the amount of radionuclides in the secondary fluid nor the fuel defect levels, but rather on the volume of blowdown processed. Therefore, the magnitude of a primary-to-secondary leak limits operations only to the extent that the leak increases blowdown rates and the quantity of chemicals to secondary water.

Operation of the system is considered normal if a total combined blowdown flow, obtained from the normal blowdown connection on each steam generator, is 0 to 155,620 (Unit 1)/186,000 (Unit 2) pounds per hour; the temperature at the blowdown heat exchanger outlet is 130°F; the condensate polisher is in operation; and the condenser tube leakage is 0 to 2 gpm.

The design bases of the system are as follows: there is a total combined blowdown flow of 308,704 pounds per hour, obtained from the normal and supplemental connections on each steam generator; the temperature at the blowdown heat exchanger outlet is 130°F; and the condensate polisher is in operation.

All materials of construction for the filters are capable of withstanding a total radiation dosage of 4×10^7 rads.

The secondary chemistry water quality will meet the specifications of the EPRI guidelines, except where CPNPP evaluations justify exceptions. Water Chemistry is addressed further in section 10.3.5.

The part of the system from the steam generator to the automatic blowdown isolation valves, outside of the Containment, is an extension of the steam generator boundary. This portion of the system is designated Safety Class 2 because it is necessary for the safe shutdown of the plant. It is also designated as seismic Category I. The portion of the system downstream of the blowdown isolation valves is not essential for safe shutdown of the nuclear plant. Therefore, these various components are classified non-nuclear-safety-related and are designed, fabricated, and tested in accordance with the ASME B&PV Code, Section VIII.

10.4.8.1.2 Seismic and Quality Group Classification

Seismic and quality group classification of the SGBS is in accordance with Section 3.2.

10.4.8.2 System Description and Operation

10.4.8.2.1 System Operation

The SGBS process flow diagram is shown on Figure 10.4-10. The physical layout of the system is shown on Figures 1.2-31 and 33. The system consists of a heat exchanger, a pressure-reducing valve, two inlet filters, two cation demineralizers, two mixed-bed demineralizers, one steam generator blowdown spent resin storage tank, one steam generator blowdown spent resin sluice pump, one steam generator blowdown spent resin sluice pump filter, and the necessary instrumentation and controls. Depending on the mode of operation, all or some of the components are placed in service. The spent resin part of the SGBS is common to both units.

1. Demineralizers in Service

During normal operation, continuous blowdown from each steam generator is routed to a common line outside the Containment. The total blowdown flows through the tube side of the heat exchanger and, after pressure reduction, through the filters and demineralizers. The cooled, demineralized, low-pressure blowdown then mixes with condensate; the combined flow is used as the coolant on the shell side of the heat exchanger. The coolant, which is now at a higher temperature because of the heat picked up from the hot blowdown, flows to the heater drain tank.

During higher levels of condenser tube leakage, supplemental blowdown can be obtained by opening the normally closed supplemental blowdown connection on the steam generator. This supplemental flow and normal blowdown flow from each steam generator are routed together and are processed as described for the normal mode of operation. The routing of the blowdown lines within the Containment is shown on Figure 10.3-1. Blowdown from each steam generator flows into a common header. Each line has a Containment isolation valve (HV-2397, HV-2398, HV-2399, and HV-2400), a steam generator HELB isolation valve (HV-2397A, HV-2398A, HV-2399A, and HV-2400A), a blowdown control valve HV-5175, HV-5176, HV-5177, and HV-5178), and a manual isolation valve (MS-151, MS-153, MS-155, and MS-149). The blowdown header is connected to a high-pressure bulk nitrogen gas supply system. The nitrogen gas is used to provide a nitrogen blanket to the steam generator during wet layups and to mix the layup chemicals.

The blowdown enters the heat exchanger at a temperature of 567°F and a pressure of 1107 psi and leaves the heat exchanger at a temperature of 130°F and a pressure of approximately 1000 psia. Flow element FE-5219 measures the total blowdown flow which, in conjunction with the blowdown water chemistry measurements (see Process Sampling System, Section 9.3.2), guides the operator in adjusting balancing valves HV-5175 through HV-5178 to control individual blowdown. Once the balancing valves are set they are locked in position and do not change. Temperature element TE-5182, located downstream of the flow measuring instruments, measures the exit temperature of the cooled blowdown. Valve TV-5182 is used to regulate the flow of condensate coolant, thus maintaining the design exit temperature.

The cooled blowdown is depressurized from 1000 psia to 300 psia before entering the filters and demineralizers. PV-5180 is a pressure-reducing throttling valve. In case of a

failure in the pressure-reducing system, the fluid pressure is relieved through pressure relief valve SB-020 to the condenser.

The cooled and depressurized fluid is then filtered. There are two filters in the system, which are normally operated in parallel in order to maximize filter life. The filters remove suspended solids. When the pressure drop across a filter reaches a predetermined value that filter is removed from service. The spent filter cartridge assembly is removed as a complete unit and is then disposed of in an appropriate manner. A new filter cartridge assembly is inserted, and the filter is then placed in service. The filtered water temperature is measured by TE-5186 and indicated on the local panel.

The filtered water is then passed through the demineralizers, comprised of four ion exchange units, along with the necessary valves and instrumentation. Manual samples can be taken from each demineralizer outlet. Any resin fines which pass through the ion exchangers are removed in the SGBS resin traps which are Y-type strainers with fine mesh screening designed to arrest resin fines. The effluent flow is measured by flow transmitter FT-5221 and is indicated on the control panel. The processed blowdown is sampled for radioactivity as described in Section 11.5.4 and Table 11.5-4. Process sampling to determine resin exhaustion and performance is described in Section 9.3.2 and Table 9.3-4. If the activity level is within acceptable limits, the fluid is recycled to the heater drain tank or the condenser. If the secondary side radioactivity level is above the acceptable limits, operator action closes valve RV-5179 and valves HV-2397 through HV-2400, thus isolating the system.

2. Demineralizers Bypassed

The demineralizers may be bypassed when the following conditions exist:

a. Activity level of the blowdown, as checked by RE-4200, in the Process Sampling System, is below the detectable level.

b. The processed blowdown is not being routed to the heater drain tank.

c. A transient chemistry condition, such as condenser in leakage, is not in effect.

Under those conditions, the blowdown is cooled, depressurized, filtered, and sent back to either the condenser (normal operation and startup), the Condensate Storage Tank (draining the steam generator for level control), or the vent and drain system of the Turbine Building (during draining of the steam generator for dry layup).

3. Resin Replacement and Removal

When an exchanger vessel is exhausted, as indicated by either high pressure loss across a vessel or high conductivity in the effluent (measured in the Process Sampling System), the spent resin is removed and replaced with fresh resin. Pressure loss across the cation vessels is measured by the difference between the vessel inlet and outlet pressure gauges (PI-5184, PI-5193, and PI-5194). Similarly, pressure differential for the mixed-bed vessels is measured using gauges PI-5195, PI-5198, and PI-5199. Conductivity is measured from the samples routed to the Process Sampling System

downstream of the cation resin trap and upstream from control valve RV-5179. High conductivity also indicates resin exhaustion.

To remove resin, one vessel is valved out of service, and the entire flow is routed to the remaining vessel. By proper alignment of valves, the SGBS spent resin sluice pump pumps water from the SGBS spent resin storage tank to the exhausted vessel through the normal outlet underdrain. Water exits the vessel from the upper screened outlet and flows back to the SGBS spent resin storage tank. This backwash loosens the resin bed.

Next, the SGBS spent resin storage tank sluice pump discharge is routed through the underdrain and the resin exits the vessel bottom. This fluidizes the resin to a 30 to 50 weight percent slurry which is routed to the SGBS spent resin storage tank. The ion exchange vessel is drained after resin is removed, and new resin is added through a flexible connection. The vessel and the fresh resin bed is then rinsed and pressurized. At this point, the vessel is ready for service.

Resin slurry is allowed to settle in the SGBS spent resin storage tank. A lateral distribution of suction screens lead to a common nozzle at the tank wall to provide suction to the SGBS spent resin sluice pump. The tank has a nitrogen gas pressurizing line at the top and six other nitrogen gas lines at the bottom for fluidizing the settled resin. By pressurizing this tank, the fluidized resin slurry is routed to the mobile processing connection.

The water pumped by the SGBS spent resin sluice pump is filtered to remove suspended solids. This filter is identical to the SGBS inlet filters.

10.4.8.2.2 Component Description

1. Steam Generator Blowdown Heat Exchanger

   One heat exchanger cools the blowdown before it is sent to the demineralizers for cleanup. The heat exchanger is of a shell and tube design with blowdown water on the tube side cooled by a combined flow of the cooled blowdown and condensate on the shell side.

2. Steam Generator Blowdown Inlet Filter

   This filter removes particulate matter from the blowdown before it flows to the demineralizers. Two filters are provided.

3. Steam Generator Blowdown Cation Demineralizers

   Two cation-bed demineralizers are provided to remove radioactive materials and other dissolved solids. The demineralizers are connected to the sluice pump and the spent resin storage tank during resin replacement.

4. Steam Generator Blowdown Mixed-Bed Demineralizers

   Two mixed-bed demineralizers, are provided downstream of the cation demineralizers to remove additional radioactive materials and dissolved solids not removed by the cation

demineralizers. The demineralizers are connected to the sluice pump and the spent resin storage tank during resin replacement.

5. Steam Generator Blowdown Spent Resin Sluice Pump

   This canned, centrifugal pump is provided to sluice the spent resin from the demineralizers to the spent resin storage tank. The pump draws water from the spent resin storage tank through a screen and directs it to the demineralizers. A resin slurry is formed and routed to the spent resin storage tank. This pump, which is common to both units, is also used to backwash and rinse the demineralizers.

6. Steam Generator Blowdown Sluice Filter

   This filter removes resin fines from the sluice water which is directed to the demineralizers. This filter is common to both units.

7. Steam Generator Blowdown Spent Resin Storage Tank

   This tank is used to collect and store the spent resin from the SGBS demineralizers until the resin is transferred for disposal. This tank is vented to the plant ventilation system; it is common to both units.

10.4.8.2.3 Equipment Design Criteria

Design criteria for the equipment are listed in Table 10.4-6.

10.4.8.3 Safety Evaluation

Process control and protective instrumentation are provided to ensure functional integrity and the rating efficiency of the system.

The SGBS does not discharge radioactivity to the environment. Whether or not the SGBS is available, plant safety is not affected. The steam generator can be operated without blowdown during the time required for SGBS maintenance. If blowdown cleanup is not available for a period of time, the secondary-side chemistry can exceed the operating limitations.

During abnormally high primary-to-secondary leakage, the operation of demineralizers, will process the blowdown for recycle to the condenser. The demineralizers become exhausted more frequently as the blowdown rate is increased. Because the loss of the SGBS is unlikely, steam generator shell-side radioactivity concentration should not exceed the values given in Table 11.1-7.

Failure of this system has no effect upon safety-related systems. A failure and effects analysis of the system components is shown in Table 10.4-7.

Materials of construction of the steam generators and the blowdown cleanup system are compatible with secondary cycle water chemistry during any primary-to-secondary leaks.

10.4.8.4 Tests and Inspections

1. Inspections

To keep radiation exposures to operating personnel ALARA, the SGBS is designed to control leakage and permit maintenance in accordance with the guidelines of NRC Regulatory Guide 8.8. Planned and carefully prepared periodic visual inspections and preventive maintenance will be conducted by experienced and trained personnel. All components are accessible for internal and external inspection and maintenance.

2. Tests

The SGBS will be tested in place initially and also following any major repair or replacement. Visual inspection of the system and all associated components will be made before each test.

10.4.9 AUXILIARY FEEDWATER SYSTEM

10.4.9.1 Design Bases

The Auxiliary Feedwater System shown on Figure 10.4-11 is designed to provide a supply of high-pressure feedwater to the secondary side of the steam generators for reactor coolant heat removal following a loss of normal feedwater. The system is used in lieu of the main feedwater during cooldown, and startup operations. It also provides a cooling source in the event of a loss-of-coolant accident (LOCA) for small breaks (see Section 15.6). Furthermore, the system is used in the event of a main steam line break, feedwater line break, Control Room evacuation, and steam generator tube rupture.

The system functions over the normal operating pressure range of the steam generators, 100 psia to 1106 psia (Unit 1) or 1107 psia (Unit 2), and is capable of supplying the minimum required flow to at least two of the effective steam generators against a back pressure equivalent to the accumulation pressure of the lowest set safety valve (1236 psia) plus the system frictional and static losses.

The Auxiliary Feedwater System is designed to preclude the effects of hydraulic instability due to water hammer by supplying water to the secondary side of the steam generator through a separate upper auxiliary feedwater nozzle. This permits the cold auxiliary feedwater to be heated as it comes down the side of the steam generator prior to reaching the feedwater preheater. For further discussion of the feedwater system upper nozzle arrangement and considerations see Subsection 10.4.7.

The water level in steam generators is maintained at the proper level to prevent a temperature rise in the RCS, which could result in the release of primary coolant through the pressurizer relief valves.

To ensure that steam does not flow back to the auxiliary feedwater nozzle, the steam generator water level should be above the auxiliary feedwater discharge pipe at all operating levels where steam formation can occur. Therefore, the operator is cautioned to maintain SG level above the auxiliary nozzle internal pipe extension when the temperature is above 212°F. This pipe is

designed with a loop seal immediately upstream of the SG nozzle to prevent steam backleakage. Forward flushing is provided through the auxiliary feedwater nozzle at all loads.

In the event that steam does flow back past the loop seal and the check valves in the feedwater line to the upper nozzle, the temperature of this line will increase causing temperature elements in the line to alarm in the control room. This allows the operator to take action to resolve the problem.

With either only onsite or only offsite power available with an assumed single failure, sufficient auxiliary feedwater flow to two steam generators is provided to permit operation at hot standby for four hours, followed by a cooldown period, at a cooldown rate of 50°F/hr, to reduce the T_{avg} to 350°F, at which time the RHR System can be operated. In the event of a main steam or feedwater line break, sufficient auxiliary feedwater is provided to permit operation at hot standby for two hours followed by the above mentioned cooldown period.

Two motor-driven pumps and one turbine-driven-pump are provided to ensure an adequate supply of auxiliary feedwater to the intact steam generators following a feedwater line break accident, coincident with the assumed single active failure. Subsequent action to provide additional auxiliary feedwater flow is described in Section 15.2.8.

All redundant components are physically separated from each other by an arrangement of concrete barriers designed to preclude coincident damage to equipment in the event of a postulated pipe rupture, equipment failure, or missile generation [6] [7].

The system has been designed to withstand the adverse environmental conditions delineated in Section 3.11, including the effects of flooding which are further discussed in Subsection 10.4.9.2, System Description.

The system is classified as nuclear-safety-related and consists of ANS Safety Class 2 and 3 piping and equipment, except for the non-nuclear-safety condensate transfer pump and associated piping and valves used to provide makeup, recirculation and drainage for the Condensate Storage Tank [3] [4]. Seismic Category I design criteria are considered for all ANS Safety Class 2 or 3 components. Seismic requirements are given in Section 3.2 [5]. The piping is designed to meet the requirements of Branch Technical Positions APCSB 3-1 and MEB 3-1. The system is designed in accordance with 10 CFR Part 50, GDC 2, 4, 5, 19, 44, 45, 46, 54, and 57 [8], [9], [10], [11], [12], [13], [14], [17], [28].

Unit 1

On Unit 1 the steam generators have been replaced with a model Delta 76 that employs a feedring design that eliminates the preheater. Auxiliary feedwater and feedwater entering the steam generator merge with the saturated liquid removed by the moister separators. The combined feed then flows down the annulus formed by the steam generator shell and the tube bundle wrapper where it enters the tube bundle.

To ensure that steam does not flow back through either the main feedwater or auxiliary feedwater nozzles, the main feedwater and auxiliary feedwater lines should be water filled at conditions where steam formation can occur. The operator is cautioned to maintain the SG level above the feedring at normal operating conditions. The normal operating Narrow Range level indication of

60% to 75% would result in sufficient water inventory to maintain the feedring and the AFW vertical perforated spray pipe submerged during the majority of expected transients like +/-10% load changes, turbine synchronizations, and plant loadings/unloadings. The supply piping to the auxiliary nozzle is designed with a loop seal immediately upstream of the SG nozzle to prevent steam backleakage.

In the event that steam does flow back past the loop seal and the check valves in the auxiliary feedwater line to the upper nozzle, the temperature of this line will increase causing temperature elements in the line to alarm in the control room. This allows the operator to take action to resolve the problem.

Unit 2

The Auxiliary Feedwater System is designed to preclude the effects of hydraulic instability due to water hammer by supplying water to the secondary side of the steam generator through a separate upper auxiliary feedwater nozzle. The Unit 2 preheater design steam generators permit the cold auxiliary feedwater to be heated as it comes down the side of the steam generator prior to reaching the feedwater preheater. For further discussion of the feedwater system upper nozzle arrangement and considerations see Subsection 10.4.7.

To ensure that steam does not flow back to the auxiliary feedwater nozzle, the steam generator water level should be above the auxiliary feedwater discharge pipe at all operating levels where steam formation can occur. Therefore, the operator is cautioned to maintain SG level above the auxiliary nozzle internal pipe extension when the temperature is above 212^{o}F. This pipe is designed with a loop seal immediately upstream of the SG nozzle to prevent steam backleakage. Forward flushing is provided through the auxiliary feedwater nozzle virtually at all loads by AFW flow at startup conditions, by preheater bypass flow during initial power ascension and by the split flow of the feedwater at RTPs above 30% full load.

In the event that steam does flow back past the loop seal and the check valves in the feedwater line to the upper nozzle, the temperature of this line will increase causing temperature elements in the line to alarm in the control room. This allows the operator to take action to resolve the problem.

10.4.9.2 System Description

The Auxiliary Feedwater System is comprised of two electric motor-driven auxiliary feedwater pumps and associated valves, piping, and controls and a third turbine-driven auxiliary feedwater pump with associated valves, piping, and controls, which is independent of the electrical power supply to the motor-driven pumps. Three pumps are necessary to ensure an adequate supply of auxiliary feedwater following an accident, coincident with the single failure of a pump. The design parameters of the auxiliary feedwater pumps are given in Table 10.4-8.

All three pumps normally draw suction from the Nuclear Safety Class 3 Condensate Storage Tank. A single line supplies water through a common locked-open valve to the suction of the motor-driven auxiliary feedwater pumps, and a second line supplies water to the suction for the turbine driven auxiliary feedwater pump. Of the approximately 500,000-gal capacity, approximately 282,000 gal remain for auxiliary feedwater. The rest of the tank is used as condensate storage for the Condensate System. The reserved auxiliary feedwater cannot be

drained by the non-nuclear-safety systems because of the elevation of the outlet nozzles. Piping tie-ins are provided to allow connection of equipment to process and chemically treat the inventory of the Condensate Storage Tank.

While the Condensate Storage Tank is the preferred water supply, another ANS Safety Class 3 alternate supply is provided. The Auxiliary Feedwater System has the capability to draw suction from the service water system (SWS) in the event of loss of the Condensate Storage Tank. Two normally closed, key-switch activated, motor-operated butterfly valves in the SWS and three motor-operated gate valves in the auxiliary feedwater system prevent contamination of the auxiliary feedwater by station service water. The three AFW valves are normally de-energized to prevent a fire induced hot-short from causing a mal-operation of the valve(s). This also de-energizes position indication in the Control Room. Therefore, these valves are “Locked Closed” to provide positive indication of correct valve position during normal operation. In addition, high-and low-point leakoff connections are provided between the SWS isolation valves and the motor-operated gate valves to allow detection of any station service water inleakage. An orifice has been installed in the leakoff lines to limit the leakage rate into the safeguard building.

Each motor-driven pump normally feeds two steam generators. A normally closed interconnection between the motor-driven pump discharge lines permits either pump to feed to all four steam generators. This interconnection provides the operator with the means to maintain the water level in all steam generators on a long-term basis following a LOCA by operating either motor driven pump. The motor driven pumps can be manually started or stopped from the Control Room or the hot shutdown panel. The turbine-driven pump discharge line branches into four separate lines each feeding one steam generator. The turbine-driven pump can be manually started from the Control Room or the hot shutdown panel.

Each of the lines that connects the three auxiliary feedwater pumps to the steam generators is provided with: a normally open, pneumatically operated feed regulator control valve; a flow-limiting orifice; a check valve; a motor operated isolation valve; and two manual isolation valves. Remote manual control of the feed regulator control valve is provided from the Control Room with provision for local manual operation on the hot shutdown panel. Air accumulators are provided for the pneumatically operated valves with sufficient capacity to permit remote valve closure in the event of a secondary system break where local valve operation cannot be accomplished within the required time period following the incident. The instrument air system and the air accumulators are described in Section 9.3.1.2. As described in Section 6.2.1.4.4, isolation of a faulted steam generator is an operator action required within 10 minutes after a secondary pipe break. Each air accumulaor is required to have enough air to close its control valve and recirculation valve and maintain them closed for a mission time of 30 minutes (i.e., until local manual control is established). Testing criteria is based on one close stroke of the valves and 30 minutes of air loss/leakage. The control valves are located near each AFW pump to allow isolation or local manual flow control as required. Flow control is not required for the 30 minutes following a secondary line break; however, if isolation is not required and instrument air is lost, the operator may control flow remotely using the air accumulators until instrument air can be restored or local manual flow control can be established.

The flow limiting orifices are provided to limit flow to any one steam generator to a maximum of 1380 gpm, in the event of either a main feed line break or a main steam line break inside containment.

An orifice-type flow measuring device is located in each of the auxiliary feedwater lines to indicate flow to each steam generator and to provide a means of detecting grossly uneven flow to the steam generators. Readout for these flow measuring devices is located in the Control Room and on the hot shutdown panel. To avoid the possibility of a single active failure stopping all auxiliary feedwater flow to a steam generator, there are no valves located in the common main feedwater lines.

The Auxiliary Feedwater System operates over an extended period of time following a LOCA. The two motor-driven pumps start automatically and they provide an additional means for removing core residual heat in the event of a LOCA for small breaks. During large break LOCA conditions, the system is used to maintain an adequate water level above the tubes in the steam generators to prevent primary to secondary leakage. The operator shuts down the pumps at his discretion and manually adjusts feed flow to individual steam generators.

Depending on the severity of the event, either the two motor driven pumps or all three auxiliary feedwater pumps start automatically after either a main steam line break or a feedwater line break (see Subsection 10.4.9.5). At an early stage in the accident, the operator isolates the feedwater to the affected steam generator. The system provides for the cooldown of the unaffected steam generators to prevent the RCS from being repressurized. The operator shuts down the pumps at his discretion.

After a loss of the main feedwater system (condition II event), either the two motor-driven auxiliary feedwater pumps together, the turbine-driven pump alone, or any combination are capable of providing sufficient flow to the steam generators to allow the plant to be taken to a safe shutdown condition (and remain a condition II event). The operator shuts down the pumps at his discretion.

The operation of the Auxiliary Feedwater System following a steam generator tube rupture is manually initiated. The two motor-driven pumps are started manually and are used to maintain the required water level in the steam generators as the plant is shut down. The operator identifies the affected steam generator and isolates it and the operator shuts down the pumps at his discretion.

The operation of the Auxiliary Feedwater System following a Control Room evacuation is manually initiated and is controlled from the hot shutdown panel. The operator maintains water level in the steam generators with either the two motor-driven pumps or the turbine-driven pump. The pumps are used to maintain the required water level in the steam generators as the plant is shut down. Again the operator shuts down the pumps at his discretion.

Each power supply train for the motor-driven pumps, control valves, and instrumentation is supplied from a separate and independent Class 1E bus that is capable of supplying the minimum required power for the safety-related loads required following a LOCA or loss of offsite power (blackout), or both. Each bus can be powered from two independent offsite power sources or by the diesel generator assigned to the bus [15].

The control power for the non-safety related turbine driven pump test panel is from the Train A Class 1E bus and is isolated by an “S” signal.

Unit 1

Downstream of the last isolation valve, each line from the motor-driven pumps joins with a corresponding line from the turbine-driven pump to form a common line that connects with the feedwater line that connects to the auxiliary nozzle on the Unit 1 steam generators.

Unit 2

Downstream of the last isolation valve, each line from the motor-driven pumps joins with a corresponding line from the turbine-driven pump to form a common line that connects with the feedwater preheater bypass line. The preheater bypass line connects to the auxiliary nozzle on the Unit 2 steam generators.

10.4.9.3 Safety Evaluation

The Auxiliary Feedwater System is designed to ANS Safety Class 2 and 3 requirements, and in the event of loss of offsite power, the backup turbine driven auxiliary feedwater pump operates. The turbine drive does not have any auxiliaries requiring electrical power. For redundancy, steam for the turbine driver is supplied from two steam generators. Either supply can meet the turbine driver requirements. The turbine steam supply valves are fail-open air-operated types each with a pilot solenoid valve supplied from a redundant Class 1E power supply. The Train A steam admission valve supplies steam from steam generator 4 and the Train B steam admission valve supplies steam from steam generator 1.

The turbine speed control governor is of the mechanical/hydraulic type, which is capable of maintaining the turbine at the high speed setting without any outside sources of power.

During normal plant operation the turbine speed is controlled by the speed setting signal which is converted to a pneumatic signal for the turbine governor controls. Loss of this remote speed setting signal and/or the air supply will result in the turbine running at the high speed setting.

The power supply for the turbine speed setting signal is from the station inverters which are supplied from the 125 volts dc batteries. The air supply is from the station instrument air system.

Safe shutdown of the unit relies upon the availability of the Auxiliary Feedwater System. Loads which are required for the safe shutdown of the unit are connected to the Class 1E power supply.

The auxiliary feedwater pumps are designed to deliver the minimum required flow within 60 seconds (including sensor/signal response time). The 60 second design response time for the Turbine Driven Auxiliary Feedwater pump is less than that evaluated in the accident analyses. A maximum response time of 85 seconds from the Steam Generator LO-LO setpoint (including sensor/signal response time) for the Turbine Driven Auxiliary Feedwater pump to deliver the minimum required flow is required based on the accident analysis assumptions.

In the event of a LOCA or loss of all offsite power (blackout), or both, the motor-driven auxiliary feedwater pumps and their associated motor-operated valves are automatically sequenced onto their respective emergency buses.

Motor-operated valves stop automatically when valve action is completed while the motor-driven auxiliary feedwater pumps must be manually stopped.

A failure mode analysis (electrical and mechanical) is shown in Table 10.4-9 and on Figure 10.4-12, respectively.

In the event of a feedwater line break inside the Containment, the larger than normal flow is detected by the flow-measuring device in the line.

Sufficient redundancy is provided throughout the Auxiliary Feedwater System and supporting systems to ensure safe plant shutdown with only one motor-driven auxiliary pump by supplying the required flow to a minimum of two steam generators while subject to a single active failure in the short-term or a single active or passive failure in the long-term. This flow is sufficient to maintain the unit in a safe condition.

The Auxiliary Feedwater System is capable of withstanding adverse environmental conditions. It is designed to seismic Category I requirements, is located within tornado resisting structures, and is protected from tornado-generated missiles. The Condensate Storage Tank is designed against tornadoes and missile penetration. The supply lines from the tank to the Safeguards Building are located in seismic Category I structures and the auxiliary feedwater pumps are located in an enclosed bay of the Safeguards Building at a floor elevation of 790 ft 6 in.

All redundant components (including pumps, controls, Class 1E power sources, and electric cable) are separated from each other by a proper arrangement of barriers or suitable physical separation. This barrier separation is provided to preclude coincident damage to redundant equipment in the event of a postulated pipe rupture, equipment failure, or missile generation. Each pump is situated in a separate compartment and is protected by walls constructed to seismic Category I requirements.

Two sources of flooding are considered. One source of flooding is a pipe break in the auxiliary feedwater pump discharge. Separate compartment design, as well as access and drainage, prevents flooding of adjacent equipment. The second source of flooding considered is a crack in a 24-in. diameter 0.375-in. wall, component cooling water pipe that runs adjacent to the bay occupied by the auxiliary feedwater pumps. This piping is considered a piping system containing moderate-energy fluids during reactor operation. Floor drains are provided to accommodate any water leakage as a result of a postulated crack.

Redundancy of cooling water source is ensured by a connection with the SWS, which is of Safety Class 3 design.

This backup source of water, which has lower quality standards than those specified for steam generator feed, would be used only in case of extreme emergency, when safety overrides water quality consideration. The required capacity and design conditions for the use of the SWS and SSI as an alternate feedwater source are described in Section 9.2.5, and show that the SSI is able to supply water to the steam generators for the required time.

For Unit 2, design of the Auxiliary Feedwater System is such that the effects of water hammer are precluded by the use of a separate upper auxiliary feedwater nozzle on the steam generator. (See Subsection 10.4.7 for further discussion.)

10.4.9.4 Inspection and Testing Requirements

All system components are tested and inspected in accordance with the applicable codes. The system is capable of being tested while the plant is in operation. A test line to the Condensate Storage Tank is provided on each pump discharge. This provision allows each pump discharge valve to be closed. Each pump can be started manually and recirculated back to the tank. Only one pump at a time is tested and pressure and flow indications at the pump discharge are used for checking the pump performance.

Refer to Section 6.2.4 and 6.2.6 for containment isolation valve testing.

10.4.9.5 Instrumentation Requirements

1. General

The instrumentation and controls for the Auxiliary Feedwater System provide for automatic or manual and remote or local operation of the system. Controls for manual operation of the system at local stations and at the Hot Shutdown Panel are provided in addition to auto/manual controls in the Control Room. Controls from the Hot Shutdown Panel override all other signals and activate an override alarm in the Control Room.

For a description of the Auxiliary Feedwater System instrumentation and controls, refer to Section 7.3.1.1.4, item 5.

Automatic Initiation of the AFW System will automatically isolate steam generator blowdown and sampling for all steam generators.

2. Feedwater Flow Control

During cooldown, the operator maintains the required steam generator water level by varying the auxiliary feedwater flow. Motor-driven auxiliary feedwater pump flow to each steam generator is remote manually controlled by feed regulator control valves. The valves are located near the pumps to allow manual operation. Control stations on the Main Control Board and Hot Shutdown Panel enable the operator to control the flow manually from the Control Room or from the Hot Shutdown Panel in conjunction with a nearby patch panel for valve control.

An automatic motor driven auxiliary feedwater pump start signal automatically trips the manual flow control to automatic full open to ensure flow to the steam generators. Motor driven auxiliary feedwater pump runout protection is provided by the system piping configuration, and flow restricting orifices. Each auxiliary feedwater regulator control valve is air-operated and is provided with a nuclear safety-related air accumulator to permit valves to close in the event of a secondary system break and an instrument air system failure. The valves fail open on loss of air or electric failure.

All controls for motor-driven pump A are electrical Train A oriented; all controls for motor-driven pump B are electrical Train B oriented; controls for the turbine-driven pump are fed from the Train A, Class 1E 125VDC System.

On loss of electrical power or air supply, the turbine-driven pump accelerates to maximum speed demand. Since the turbine-driven pump is supplied with a fail-closed trip and throttle valve, this valve is latched in the open position. Two redundant steam supply lines, each with an air operated supply valve, provide steam to start and accelerate the turbine-driven pump. These air-operated valves fail open, ensuring that the pump starts on loss of air supply or electrical power. Speed control is accomplished with a Woodward mechanical/hydraulic type governor. A mechanical overspeed trip device is provided to trip the turbine at 116.6-percent rated speed. Manual speed control is from the Control Room or the Hot Shutdown Panel. The manual control from the Hot Shutdown Panel overrides all other signals. There is speed indication on the Control Room Panel and Hot Shutdown Panel and at the local panel. Flow from the turbine-driven pump to each steam generator is regulated by control valves under manual control from the Control Room, the Hot Shutdown Panel, or locally. Each valve has an air accumulator to permit the valves to close in the event of a secondary system break and an air system failure.

3. Feedwater Supply Control

Filling of the Condensate Storage Tank may be performed manually by the operator aligning demineralized water, or automatically by associated level instrumentation when the demineralized water source is continuously aligned. Tank level is indicated locally and remotely and HI-HI, LO, and LO-LO tank level alarms are provided. Redundant level transmitters are used.

The Condensate Storage Tank supplies water to the auxiliary feedwater pumps. The automatic starting of any auxiliary feedwater pump initiates the automatic isolation of the Condensate Storage Tank from all its other users. This maximizes the water supply to the auxiliary feedwater pumps whenever they are started.

The condensate transfer pump is manually started and stopped from a main control board switch. The pump is automatically stopped in the event of an “S” signal or on low pump suction pressure.

4. Emergency Feedwater Supply Control

Inlet motorized control valves are manually controlled by a key lock switch to admit service water to the suction of the auxiliary feedwater pumps.

5. Display Information, Alarms, and Controls

Control switches and position indication lights are provided for all remotely operated valves.

The following display information and alarms, in addition to those already mentioned, are provided in the Control Room:

a. Suction pressure indication and low alarm for each auxiliary feedwater pump

b. Temperature indication for each steam generator auxiliary feedwater line

c. Low pressure alarm for alternate feed supply from service water system (Unit 1 only).

d. Discharge pressure indication for each auxiliary feedwater pump discharge; pressure indication on hot shutdown panel for these pressures; low discharge pressure alarm for each motor driven auxiliary feedwater pump

e. Flow in the discharge line to each steam generator; indication is also on the hot shutdown panel.

f. Flow in the discharge line from each pump

g. Alarms for local override control from the hot shutdown panel; local indicators for temperature, pressure, flow, and level are provided as shown on the flow diagram.

h. Alarms for overspeed trip, lube oil system trouble and high drip pot level for AF pump turbine.

10.4.10 EXTRACTION STEAM SYSTEM

The primary function of the Extraction Steam System is to convey the steam required for regenerative heating of condensate and feedwater by the feedwater heaters. The extraction system also conveys auxiliary steam for process use in the NSSS.

10.4.10.1 Design Bases

The Extraction Steam System design meets the requirements of the following codes and standards:

1. American National Standards Institute (ANSI)

2. American Society of Mechanical Engineers (ASME)

3. American Society for Testing and Materials (ASTM)

4. American Welding Society (AWS)

5. Heat Exchange Institute (HEI)

6. Manufacturer's Standardization Society (MSS)

7. Occupational Safety and Health Act (OSHA)

8. Steel Structures Painting Council (SSPC)

9. Tubular Exchanger Manufacturers Association (TEMA)

The system design parameters are based on the turbine cycle heat balances shown in Section 10.1.

10.4.10.2 System Description

The Extraction Steam System and interconnected piping is shown on Figure 10.4-13. The extraction steam lines to the different heaters are separated into two groups in order to prevent water induction into the turbine, turbine overspeed, and line drainage. One group comprises extraction lines to two high pressure heaters and the other group includes extraction lines to four low pressure heaters. The heaters are numbered 1 through 6 with No. 1 being the highest pressure heater. The heaters are in two parallel strings and are differentiated as 1A, 1B, 2A, 2B and so forth.

Each extraction line, with the exception of heater Nos. 5 and 6, is provided with drain pots to remove condensate to the condenser. The extraction lines to heaters 1, 2, 3 and 4 are each equipped with a motorized stop valve and power-assisted non-return check valve.

The power-assisted check valve, because of its fast closing time, provides adequate protection for the turbine against the possibilities of overspeed caused by steam which may flow back into the turbine in the event of a turbine trip.

The motor-operated shutoff valve is a slower closing valve, but provides protection to the turbine against the possibility of water induction when there is a high water level in the feedwater heater.

The stop valves can be manually operated from the Control Room. The drain valves can also be operated manually during startup and periods of light load operation. Each of the air-operated drain valves, power-operated check valves, and motorized stop valves has position-indicating lights in the Control Room.

Heaters 5 and 6 are located in the condenser neck with the channel ends extended out of the condenser. Check valve and motorized stop valve combinations are not provided on the extraction lines because of anti-flash baffles to restrict the reverse flow from these heaters to a sufficiently low flow so that it cannot adversely affect turbine overspeed. These lines do not contain enough available energy to overspeed the turbine.

To prevent water induction into the low-pressure turbines from low-pressure heaters 5 and 6 as a result of tube rupture, high-high water level switches on the heaters transmit signals to close the motorized valve located on the discharge line of the condensate pump thereby preventing the flow into the heaters.

In addition, the Extraction Steam System also provides steam for process use.

On turbine trip, pressure transmitters from the turbine provide a signal to close all the power-operated check valves and the motorized stop valves.

Safety relief valves have been provided on the shell side of the high-pressure heaters and the low-pressure heaters, with the exception of condenser neck heaters, to protect against overpressurization. To conform with the standards of the Heat Exchange Institute, the safety relief valves are sized to pass a minimum of 10 percent of the tube side flow.

Extraction steam lines from any particular extraction point are connected by a pressure equalizing header before the lines enter the respective heaters. Pressure equalizing headers are not provided for heaters 5 and 6 because each of the same numbered heaters receives

extraction steam from different low pressure turbines and the extraction steam lines from these turbines are contained in their respective condensers.

All heaters, piping, valves, and miscellaneous components of the Extraction Steam System with the exception of condenser neck piping and heaters 5 and 6 that have a surface temperature of 125°F or more are insulated to minimize heat loss and afford personnel protection.

In order to provide adequate piping flexibility and to keep loads and moments on the turbine extraction nozzles within acceptable limits, all the extraction lines from the low pressure turbine are provided with flexible expansion joints located inside the condenser neck.

10.4.10.3 Safety Evaluation

The system is non-nuclear-safety-related, and seismic design criteria have not been considered.

10.4.10.4 Tests and Inspections

All system components are tested and inspected in accordance with the applicable codes.

10.4.10.5 Instrumentation Requirements

Each turbine side drain pot is provided with an extraction line drain valve. These valves are opened and closed manually with control-board-mounted switches. After a valve opens, it remains open until manually closed. The valves are air operated and fail open on loss of electrical signal or air supply. These valves are automatically opened to drain lines when the extraction stop valve in the extraction line closes.

High water level in the heater side drain pots, which are located between stop valves and heaters, alarms in the Control Room.

Extraction line positive closing check valves for heaters 1, 2, 3 and 4 are automatically closed by high-high level in the associated heater, or a turbine trip (pressure signal from turbine oil control system). After automatic closure, valves must be opened manually by control board switches. They are not capable of being closed manually. In addition, high-high levels in the heaters and drain pots trigger alarm signals.

The extraction line motorized stop valves, like positive closing check valves, close on a high-high heater level or a turbine trip. Manual opening and closing from a control board switch is possible. Each stop valve has its own control board switch with three positions: close, auto, and open. Switches spring return to auto. Extraction line drain valves are interlocked to open when extraction line stop valves close.

Heaters 5 and 6, in the condenser necks, do not have extraction shutoff valves.

The motorized stop valve in the auxiliary steam supply extraction line is automatically closed by a high-high level signal from the extraction line drain pot level switch or by turbine trip.

Heaters 1 thru 4 have a pressure transmitter in their extraction lines to transmit signals to the control board pressure indicator. Heaters 1 through 4 have local pressure indicators on the turbine side of extraction line stop valves. These are used when a heater is put back in service

after being taken out to ensure that heater shell pressure is lower than corresponding turbine stage pressure before the stop valve is opened.

All drain pot level switches alarm in the Control Room on high-high water level. All air-operated drain pot drain valves have open-close indication lights on the control board and they alarm when opened.

Extraction line stop valves and positive closing check valves also have open-close indicator lights on the control board.

10.4.11 HEATER DRAINS SYSTEM

This section discusses the Heater Drains System in accordance with the requirements of NRC Regulatory Guide 1.70 [16].

10.4.11.1 Design Bases

The Heater Drains System functions to aid in regeneratively heating feedwater by cascading the higher energy drains through successively lower energy stages of feedwater heaters. The system also returns the saturated water from the extraction steam to the Feedwater System and the Condensate System in a manner which results in good cycle efficiency.

The Heater Drains System is designed to perform the following functions:

1. To drain the moisture removed from the moisture separator/reheater (MSR) shell and separator sides to the MSR shell and separator drain tanks from where it is continuously drained to the feedwater heater drain tank 01.

2. To collect the steam condensed in the reheater tubes of the MSR in the reheater drain tanks from where the saturated water is continuously drained to the high-pressure feedwater heaters

3. To cascade the higher energy drains through successively lower stages of feedwater heaters to provide regenerative heating for feedwater before its admission to the steam generators

4. To collect drains in the feedwater heater drain tanks from which the heater drain pumps take their suction and inject the condensate into the Feedwater System

5. To provide the alternate drain paths for condensate flow during emergency conditions

The Heater Drains System also functions to remove the noncondensable gases from the system before gas accumulation to improve the heat transfer performance.

The Heater Drains System also incorporates a controlled cold water injection (condensate pump discharge) to the suction side of the heater drain pumps. The Condensate System also supplies seal water injection for the heater drain pumps.

All components of the Heater Drains System are designed in accordance with the applicable requirements of the following codes:

1. American National Standards Institute (ANSI)
2. American Society of Mechanical Engineers (ASME)
3. American Society for Testing and Materials (ASTM)
4. American Welding Society (AWS)
5. Heat Exchange Institute (HEI)
6. Hydraulic Institute (HI)
7. Institute of Electrical and Electronics Engineers (IEEE)
8. Manufacturer's Standardization Society (MSS)
9. National Electrical Code (NEC)
10. National Fire Protection Association (NFPA)
11. Occupational Safety and Health Act (OSHA)
12. Steel Structures Painting Council (SSPC)
13. Tubular Exchanger Manufacturers Association (TEMA)

The Heater Drains System piping is designed in accordance with ANSI B31.1; the piping is insulated where exposed to the weather.

The design bases of the Heater Drains System are as shown on the turbine cycle heat balances in Section 10.1.

10.4.11.2 System Description

10.4.11.2.1 General

The secondary plant steam turbine power cycle uses a regenerative feedwater heating arrangement consisting of two parallel strings of two stages of high-pressure feedwater heaters in series and four stages of low-pressure feedwater heaters. For each heater, except for the heater No. 3 and heater No. 6, an integral drain cooler is provided. Heater No. 6 incorporates an external drain cooler. The feedwater heaters are numbered from 1 through 6 with No. 1 heater being the highest pressure heater. Since the heaters are in two parallel strings, they are further differentiated as 1A and 1B, and so forth. In addition, two MSRs are provided to separate the moisture from steam exhausted from the high-pressure turbine and reheat it before its admission to the low-pressure turbine cylinders.

The system is provided with all the necessary instrumentation, controls, piping, and valves.

The Heater Drains System and interconnected piping are shown on Figure 10.4-14.

All drain flows from Feedwater Heaters 2A, 2B, 3A and 3B are directed to Heater Drain Tank 2. All level control instrumentation is connected to Heater Drain Tank 2 from which the heater drain pumps take suction. All other MSR drains and the Steam Generator Blowdown Heat Exchanger cooling water flow are directed to Heater Drain Tank 1 which is interconnected to Heater Drain Tank 2 through an equalization vent header between the tanks' vapor spaces and an equalization heater between the tanks' liquid spaces.

The heater drain pump recirculation flow is routed to the heater drain tank equalizing liquid space line.

10.4.11.2.2 Moisture Separator/Reheater

For each unit, two MSR assemblies are provided and located in the crossover/crossunder piping system between the high-pressure and the low-pressure turbines. The MSR are housed in one pressure vessel.

The drainage of the MSR is accomplished by the heater drains system using separate piping and drain tanks.

The drain from the MSR is by gravity. To establish a steam water interface, MSR shell, separator, and reheater drain tanks are provided. The drain tanks are of the horizontal, cylindrical with spherical-head type.

10.4.11.2.3 Feedwater Heaters

The feedwater heaters are designed to maintain a drainage flow within the limits required for safe, continuous plant operation under constant or fluctuating load conditions.

All feedwater heaters with the exception of heater No. 3 and No. 6 are equipped with an integral drain cooler. An external drain cooler is provided for heater No. 6. On its way through the drain cooler, the steam condensed is subcooled, providing an additional heating for the feedwater and improving the cycle thermal efficiency.

10.4.11.2.4 Heater Drain Pumps

Two 50-percent-capacity, motor-driven, constant-speed, horizontal centrifugal pumps are provided for the Heater Drains System. Each heater drain pump is equipped in the suction line with a gear operated, wafer-body, butterfly valve for pump isolation, with an expansion joint for thermal and vibration movements, and with an in-line temporary strainer for use during system start-up. The discharge line is equipped with a check and a gear-operated manual valve in series and a warmup line for instant startup service when only one pump is required to operate and the other is on standby.

In addition, each heater drain pump incorporates a minimum recirculation control system and a common cold water injection line.

10.4.11.3 Safety Evaluation

The Heater Drains System is non-nuclear safety-related, and seismic design criteria have not been considered.

Each primary flow path of the MSR drain tanks is provided with a check valve to prevent backflow of flashing steam from interfering with drainage to the condenser when pressure is reduced during reduction of load.

The MSRs and all heaters except Nos. 5 and 6 have steam side safety valves.

Each MSR vessel is provided with a pressure transmitter which lights an indicator in the Control Room to indicate to the operator the high MSR vessel pressure.

10.4.11.4 Tests and Inspections

Each feedwater heater, external drain cooler, pump, drain tank, MSR shell and tube, and valve receives a shop hydrostatic test performed in accordance with applicable codes.

Prior to initial operation, all foreign materials and oxides are removed from the piping, and the system is hydrostatically tested to confirm leaktightness. Visual inspection of pipe weld joints confirms the exterior condition of the weld.

10.4.11.5 Instrumentation and Controls

Each of the MSR drain tanks with the exception of the reheater drain tanks is provided with one level transmitter/ controller to position the control valves in order to maintain the normal water level of the drain tank. Each reheater drain tank is provided with one level/transmitter/controller that uses a split range.

The normal water level of the feedwater heaters is accomplished by their level controller, except heater No. 3, since this heater does not have an integral drain cooler and it drains directly to the heater drain tanks. Heater No. 6 water level is controlled by the level controller located at the discharge piping to its drain cooler.

The normal level control valves in the Heater Drains System have a fail closed position and the high level control valves have a fail-open position.

All primary and alternate drain valves of the feedwater heaters operate on a pneumatic signal from their associated level controller/ transmitters. The valves are of the air-operated globe type except for ball valves on the low-pressure heater drains.

Similar level control valves with controllers are used for the MSR shell and drain tanks. The primary and the alternate drain valves are of the air-operated type, modulated with a 4 to 20 mA electrical signal from an electronic level transmitter on the drain tanks. Each valve can be controlled manually from the manual auto station on the control board.

Heater Nos. 5 and 6 are provided with a high-high level alarm in the Control Room.

The Heater Drains System is equipped with level gauges for local level indication.

Heater Nos. 1, 2, 4 and 5 are provided with level switches actuated by high-and low-level signals from the level transmitters for high and low alarms in the Control Room. The heater No. 6 and heater No. 3 are provided with level switches to actuate a high-level alarm.

Level control signals are originated from heater drain tank 2 only. Each heater drain tank incorporates a high-level alarm actuated by a high-level signal from the drain tank level transmitter. Heater drain tank 2 incorporates a low-level alarm actuated by a low-level signal from the drain tank level transmitter.

The MSR drain tanks are provided with level switches for high alarms in the Control Room.

The Heater Drains System also incorporates the following monitors in the Control Room:

1. Pressure indicator for MSR

2. Drain temperature detector for computer input

3. Open and close position indicator lights for all control valves

4. Differential pressure indicator switch for in line strainers

10.4.12 TURBINE PLANT COOLING WATER SYSTEM

The Turbine Plant Cooling Water (TPCW) System is a closed-loop system designed to cool all turbine-generator-associated equipment and conventional plant equipment and systems which are classified as non-nuclear-safety-related [4]. The system acts to remove residual heat from all turbine plant associated equipment by means of the turbine-plant-cooling water heat exchangers and rejects this heat to the Circulating Water System, which acts as a heat sink.

System components are located in the Turbine Building.

10.4.12.1 Design Bases

The TPCW System is designed to remove heat from the turbine generator auxiliary systems and from various equipment of the secondary plant. The system provides corrosion-inhibited demineralized cooling water to components that can be adversely affected by direct reservoir water cooling because of corrosion or fouling. It is also designed to supply each component with the proper flow rate of cooling water (see Table 10.4-10) at a maximum cooling temperature of 107°F. The cooling water flow rate through each component can be adjusted by means of either butterfly or globe valves on both the inlet and outlet of the individual components. The water exits each component up to 118°F, whereupon it is directed back to the TPCW heat exchanger. The excess heat is removed by circulating water that flows through the tube side of the heat exchanger at a design inlet temperature of 102°F and a design tube velocity of 5 ft/sec. At a temperature approximately 12°F above the inlet temperature, the circulating water is then directed back to the SCR, where the excess heat is dissipated.

The anticipated analyses of the TPCW (through the shell side) and the circulating water (through the tube side) are shown in Tables 10.4-11 and 10.4-12, respectively.

Based on this analysis, calculation of both stability and saturation indices indicates that the water tends to form scale, which will be cleaned during extended outages. The water on the tube side of the heat exchanger has a corrosion control and dispersant chemical added. A solution of sodium hypochlorite or sodium hypochlorite and sodium bromide is added to the water to control organic and biological growth in the cooling system.

10.4.12.2 System Description

The TPCW System, schematically shown on Figure 10.4-15, is provided with one full-capacity heat exchanger and two full-capacity pumps for each unit. The pumps are of the horizontal, centrifugal single-stage type and are designed to deliver 20,000 gpm of cooling water at a design total dynamic head (TDH) of 100 feet. All pumps are connected to a common suction header and discharge manifolds. Either pump may serve as the lead pump.

The full-capacity heat exchanger utilizes circulating water on the tube side as the coolant with turbine plant cooling water on the shell side.

Each component cooled by the TPCW System is provided with either globe valves or butterfly valves on the inlet and outlet of each component. These serve a dual purpose: the valves can be used for the initial regulation of cooling flow (throttling) to provide overall system flow balancing and equalization of pressure drops, and they can be used for isolation and shutoff in the event of an individual component malfunction.

Each system is also provided with one TPCW head tank, which is located at the system's highest point and is connected to the pump suction. The tank contains demineralized water to which a suitable corrosion inhibitor has been added.

The tank is used as an initial system-filling reservoir and as a makeup water reservoir during system operation; it also provides an adequate pump net positive suction head (NPSH).

10.4.12.3 Safety Evaluation

The TPCW System is not required to maintain plant safety.

Seismic design criteria are not applicable.

Safe shutdown of the units does not rely on the TPCW System. Hence, there is no provision to provide power to this system from the emergency diesel generators.

10.4.12.4 Tests and Inspection

The TPCW head tank, heat exchanger, and associated piping are hydrostatically tested by the manufacturer prior to installation. The control system is also tested by the manufacturer to ensure proper valve and pump operation sequences. After installation, the system is again hydrostatically tested.

10.4.12.5 Instrumentation Requirements

10.4.12.5.1 Indication and Monitoring

1. The TPCW pump inlet and outlet pressures and the head tank level are locally observed and monitored. The pump suction and discharge lines are provided with local pressure indicators. Suction low pressure, discharge high pressure, and head tank low and high-high levels are alarmed, and a low-low level signal stops the TPCW pumps; low- and high-level signals control the head tank level.

2. The total cooling water supply to all coolers is measured, indicated on the control board, and high and low flows are alarmed.

   The temperature of the main supply cooling water is indicated on the control board. A resistance temperature detector (RTD) type temperature sensor is applied.

   The main cooling water return header is provided with similar temperature measurement as well as with local pressure measurement. The heat exchanger inlet and outlet are provided with local temperature indicators, all of which are bimetal thermometers.

3. Each return cooling water header of similar cooler group is provided with a local temperature indicator and (except the instrument air compressors) a local cell-type flow indicator. Two additional local differential pressure cell-type flow indicators are placed in the two main return cooling water headers; these measure the overall return cooling water flow from two groups of cooling units.

10.4.12.5.2 Control and Interlocking

1. The lead TPCW pump is started manually and is stopped by the head tank low-low level signal.

   The backup standby pump can be started and stopped by manual overriding or started automatically by a trip signal from its respective lead pump and is stopped by the head tank low-low level signal.

   The pump trip signal or auto stop signal is alarmed to alert an operator of any abnormal pump condition.

2. The cooling water head tank water supply valve is automatically opened and closed by low- and high-level head tank signals, respectively. The valve is of the air-operated, fail close type.

3. The turbine plant cooling water flow from the electrohydraulic control fluid coolers is temperature controlled by throttling an air-operated control valve. The temperature signal is supplied by the turbine manufacturer. An auto-manual bias control station is a part of this control loop.

   A similar control is used for the common cooling water of the three-unit turbine lubrication oil cooler.

4. The return cooling water header from the main generator primary water coolers and the eight-unit hydrogen coolers are provided with electric-motor-operated throttling-type control valves. Both of the valves' control signals are supplied from temperature control circuits designed and furnished by the turbine manufacturer.

10.4.13 AUXILIARY STEAM SYSTEM

This section discusses the Auxiliary Steam System in accordance with the requirements of NRC Regulatory Guide 1.70 [16].

The Auxiliary Steam System shown on Figure 10.4-16 is designed to provide a supply of low pressure steam to, and collect condensate from, miscellaneous warmup and process services in the Turbine and Auxiliary buildings. The system serves to conserve condensed steam for reuse. The system also minimizes possible leakage of radioactive contamination into the Unit 1 Turbine Building Sump #2 in case of steam coil leakage within the evaporators.

The system is non-nuclear safety-related. Seismic design criteria are not considered.

10.4.13.1 Design Bases

The Auxiliary Steam System is designed to meet the requirements listed in Table 10.4-13.

Drip pots are provided to collect the auxiliary steam condensate. Lines connect the drip pots to the main condenser and return the condensate to the condensate system.

Individual NSSS components which require auxiliary steam and which discharge condensate into the auxiliary steam drain tank are shown in Table 10.4-14.

Make-up water for the auxiliary boiler is demineralized deaerated water pumped from the Unit 1 and 2 condensate system.

10.4.13.2 System Description

10.4.13.2.1 General

The Auxiliary Steam System supplies steam at 50 and 150 psig to the distribution points shown in Table 10.4-13. This system also collects condensate from individual components listed in Table 10.4-14.

The Auxiliary Steam System consists of an auxiliary steam header (from which the steam is piped to the different points of application), an auxiliary steam drain tank, two auxiliary steam drain tank pumps, an Auxiliary boiler (provided with two boiler feedwater transfer pumps, a deaerating tank, two boiler feedwater pumps, a chemical feed system, an oil storage tank, and two oil feed pumps), and the associated pipings, valves, and instrumentation and controls. This arrangement is shown on Figure 10.4-16.

10.4.13.2.2 Startup

Prior to the initial startup of Unit 1, the electric auxiliary boiler was the only source of auxiliary steam. The original electric Auxiliary Boiler is not available for use and has been replaced by a fuel oil fired Auxiliary Steam Boiler.

The auxiliary boiler steam is initially utilized to clean the major pipelines. Steam from the auxiliary boiler or the other unit in operation is used for preheating the hot well via the sparger, for preheating the moisture separator/reheater tube sheets, and for sealing the main and feedwater pump turbine glands. Once the plant is in operation, the auxiliary boiler may be shut down and steam may be furnished by the normal supply sources.

10.4.13.2.3 Normal Operation

There are two normal operation supply sources for the Auxiliary Steam System - the main steam and extraction steam systems. These can be furnished by either one or both units.

High-pressure main steam is supplied to the Auxiliary Steam System via the main steam inlet header until it is reduced to 150 psig by means of a pressure-regulating control valve. The extraction steam supply serves as the low-pressure steam supply source and is reduced to 50 psig by means of a pressure control valve. The 50-psig steam serves as the normal supply of process steam for the three evaporator packages and the volume control system boric acid batching tank.

Figure 10.4-16 shows the path for supply of the steam to the floor drain waste evaporator package, waste evaporator package (liquid waste management system), recycle evaporator package (boron recycle system), and boric acid batching tank located in the Auxiliary Building.

The flow rate of steam to each evaporator is varied to maintain temperature or pressure set points in each evaporator by means of control valves.

Individual condensate lines, with differing pressures, are connected to a common header before the condensate flows pass into the condensate cooler, which is designed to cool the maximum flow of condensate down to 150°F at atmospheric pressure. A loop seal is provided between the condensate cooler and the drain tank. A bleed-off connection from the loop seal passes a small percentage of the total flow through a sample line before discharging into the auxiliary steam drain tank. Individual condensate lines are also connected to the sample line for leak detection purposes. A diaphragm valve on each bleed-off line is provided to ensure positive closure of the sample line. In normal operation, the bleed-off valve from the inlet header to the sample line remains open, while the other bleed-off valves remain closed.

Upon detection of radioactivity in the sample, the operator manually closes the bleed-off valve from the inlet header to the radiation monitor and opens the individual bleed-off valves one by one, in order to determine the source of the leakage.

The level-controlled drain tank is provided with an overflow line, a drain line for maintenance, a connection from demineralized water for washing, and a discharge connection to two drain pumps. These pumps are sized for 100-percent-capacity each. One pump serves as standby. Discharge of the pumps is directed to the Unit 1 Turbine Building Sump #2.

10.4.13.2.4 Auxiliary Boiler

The auxiliary boiler is designed to supply steam at a pressure of 150 psig and a rate to supply startup requirements for one unit startup. A fuel oil tank is furnished with the boiler to provide a continuous source of fuel for full load boiler operation. The fuel oil tank is bermed to prevent spillage to the environment. Fuel oil is pumped to the boiler by one of two 100% capacity auxiliary boiler fuel oil pumps. The boiler is also furnished with controls to shut itself down in the event of excess steam pressure or low, or high, water level. Feedwater is pumped to the boiler by one of two 100 percent auxiliary boiler feedwater pumps.

Intermittent boiler blowdown is required to dispose of accumulated feedwater impurities. This is accomplished by the auxiliary boiler blowdown flash tank which operates at atmospheric

pressure. Wastes are drained to the auxiliary boiler sump for final disposal to the low volume waste ponds. An open vent discharges flashed steam to the atmosphere. Blowdown to the flash tank originates from the main blowdown, and surface blowdown lines identified on Figure 10.4-16.

10.4.13.2.5 Abnormal Conditions

In the event of evaporator coil leakage, the sampling and analysis would detect radioactive contamination and the operator would shut off the steam supply control valves. Demineralized water is then introduced into the system to decrease the amount of radioactivity in the lines. All possible contaminated condensate is directed to the Liquid Waste Management System to preclude the entrance of any radioactive contamination to the Unit 1 Turbine Building Sump.

To determine the source of the leakage, individual condensate return test lines from each evaporator discharge are provided. Each test line is opened individually and the condensate is sampled until the source of the leakage is determined. Upon completion of repair, the steam supply valves are manually reopened, and normal operation of the system is resumed. An auxiliary steam sample cooler is provided to cool the condensate to approximately 100°F before it enters the sample line to protect personnel.

10.4.13.2.6 Electrical Requirements

Safe shutdown of the unit does not rely on the availability of the Auxiliary Steam System. Hence, power to this system is not provided from the standby diesel generators. Power is supplied by non-Class 1E sources.

10.4.13.3 Safety Evaluation

The amount of noncondensable gases released by the condensed steam is negligible; therefore, there is no need for the use of radioactivity monitoring throughout the system.

Drain pots are provided at the piping low points to remove any condensate in the steam lines during startup and normal operations.

Relief valves are installed downstream of every pressure regulator to relieve the system pressure in the event the regulator malfunctions. The auxiliary boiler is also equipped with safety valves set to open at maximum boiler working pressure.

10.4.13.4 Tests and Inspection

Normal preoperational tests are performed in accordance with applicable codes and standards. Initial inspection and testing are performed to ensure system integrity and completeness.

10.4.13.5 Instrumentation Requirements

10.4.13.5.1 Indication and Monitoring

Local pressure indications are provided for the auxiliary boiler feedwater transfer pump discharge, 50- and 150-psig steam headers, steam outlet from the auxiliary boiler, and the discharge side of pressure reducing control valves. The auxiliary boiler control panel, which is

part of the package, has sufficient instrumentation to control, supervise, monitor, and alarm all required system functions. A description of the radiation monitoring system can be found in Section 11.5.

10.4.13.5.2 Control

All functions of the auxiliary boiler are controlled from the auxiliary boiler control panel, except control of the two feedwater transfer pumps which are controlled from the local control panel. Auxiliary boiler fuel oil recirculation pump and chemical feed system pumps and mixers are controlled from local push button stations. All other system controls not located on the auxiliary boiler panel are controlled from the main control board in the Control Room. Self-regulated pressure control valves are provided for monitoring a constant pressure in the header.

10.4.13.5.3 Alarm

All alarms related to the auxiliary boiler are located on the auxiliary boiler control panel. A common alarm contact, which functions with the system annunciator, is provided to indicate abnormalities in the auxiliary boiler on the main control board. Alarms are also provided for high- and low-pressure in the steam headers and for high-level in the drain tank.

10.4.14 TURBINE OIL PURIFICATION SYSTEM

1. The Turbine Oil Purification System of the CPNPP is designed to continuously purify lubricating oil for the main and feedwater pump turbines. Provisions for the storage, handling, and heating of turbine oil are also included. System components are located in the Turbine Building and the yard.

2. This system is non-nuclear-safety-related. Seismic design criteria are not considered.

10.4.14.1 Design Bases

The Turbine Oil Purification System is designed to do the following:

1. To provide for the continuous removal of water and particulate contaminants in excess of three microns from the turbine oil at a flow rate of 50 gpm

2. To provide turbine oil purification without affecting the lubricating or sealing properties of the oil and without increasing the acidity or removing additives or inhibitors

3. To furnish makeup oil storage, filtration of all fresh and recoverable oil before transfer to system components, and storage for approximately 25,000 gal of oil whenever drainage of system components is required

4. To provide heating of the main oil feed to its optimum temperature

5. To provide for exhausting extracted moisture and oil vapors from system components

10.4.14.2 System Description

The Turbine Oil Purification System for each unit of the CPNPP as shown on Figure 10.4-17 consists of automatically controlled-skid mounted centrifuge purification units including local control panels (two for Unit 1 and one for Unit 2), one 300-gal sludge holdup tank, one 25,000-gal lubricating oil storage tank, one drain pump, all associated piping, valves, instrumentation, and controls, and one oil conditioner filtration unit. System components are located in the Turbine Building and the yard.

System filling is by oil tank trucks that discharge into the lubricating oil storage tank.

During normal Unit operation, the oil conditioner is primarily used. Oil flows from the main and FWP turbine oil tank discharge line into the oil conditioner. The oil flows through the precipitation compartment and into the filtration compartment before entering the storage compartment. From there the oil flows into the circulating pump which discharges to the polishing filter and then back to the main and FWP turbine oil tanks. If the oil conditioner is not being used the oil will flow through the centrifuge skid as described below.

During alternate system operation, turbine oil flows continuously by gravity from the main turbine and feedwater pump turbine oil tanks to the centrifuge skids. Normally, only one skid will be in operation. Purification is by centrifugal separation, with final polishing through a micronic oil filter. Clean, dry oil is then fed to the main turbine and feedwater pump turbine oil tanks. Each tank is furnished with a discharge line taking suction from an internal standpipe to prevent accidental tank drainage due to siphon effects.

Oil vapor exhausters are provided to remove air from the tanks and prevent moisture condensation. Drains from the oil vapor exhausters and mist eliminators are directed to the sludge holdup tank. All equipment drains not operating continuously are furnished with normally closed valves to prevent accidental drainage.

A 225-gpm drain pump is provided for the discharge of spent oil from the plant.

Electrical loads for this system are supplied from 480-V non-Class 1E power buses.

10.4.14.3 Safety Evaluation

The turbine oil skids and lubricating oil tank are all located in an enclosed room at elevation 778 ft 0 in. in the Turbine Building. As required by the fire codes, this room is sealed to prevent oil leakage outside the room in the event of a major tank rupture. The components supplied by the turbine generator manufacturer, as well as the main turbine oil tank, are similarly enclosed on the 803 ft 0 in. elevation. Safe shutdown of the unit does not rely on the availability of the turbine oil purification system. Therefore, there is no provision to provide power to this system from the standby diesel generators.

10.4.14.4 Test and Inspection

Normal periodic replacement of the micronic oil filters is performed when high differential pressure is noted. Cleaning of the centrifuge components will be performed on a periodic basis.

10.4.14.5 Instrumentation and Controls

The sludge holdup tank, and the lubricating oil storage tank are provided with high-level switches wired to the Control Room for alarm.

Local pressure indicators are provided on the drain pump and centrifuge skids. Local flow instrumentation is also included on the centrifuge skids.

The control and supervisory instrumentation associated with the two feedwater pump turbine oil tanks are wired to the Control Room for remote pressure indication and low-pressure alarms.

Remote high oil cooler discharge temperature alarm and indication for feedwater pump turbine are wired to the Control Room from the temperature indicator.

Two independent temperature switches control heater operation in the 90°F through 120°F design temperature range. A flow switch is also provided and is interlocked with the centrifuge skid feed pump. The interlock trips the heater if the pump stops or fails to operate for any reason, thus reducing the possibility of carbonization of the turbine oil by avoiding stagnation of oil in the heater. Heater controls and supervisory instrumentation are located in the skid mounted control panels. Each skid mounted control panel is wired to a common local control panel.

The 225-gpm drain pump is manually controlled from the common local control panel.

The continuously operating centrifuge skids are started and stopped manually from the controls located on their respective control panels.

10.4.15 NITROGEN AND HYDROGEN SUPPLY SYSTEMS

10.4.15.1 Nitrogen Supply System

10.4.15.1.1 Design Basis

The Nitrogen Supply System is designed to provide storage capacity to ensure regular replenishment of consumed nitrogen. The system is divided into two separate, permanent subsystems: the bulk storage supply and the bottle supply. Each of these subsystems has two supply manifolds, one for normal operation, the other to act as a standby source.

The design storage capacity for each supply package is based on the following:

1. Storage of 30 days is required for gases which are continuously consumed.

2. Storage of one gas charge is required for intermittent uses.

3. A safety margin of 10 to 15 percent of Items 1 and 2 is included.

The Nitrogen Supply System is designed to meet the applicable requirements of the ASME B&PV Code, Section VIII, Pressure Vessels, Part A, ANSI B31.1, Code for Pressure Piping, Power Piping, and ANSI B.31.8, Gas Transmission and Distribution Piping Systems.

Tables 10.4-16 and 10.4-17 list the components which receive nitrogen from this system. Certain components require only a one-time fill prior to reactor startup or shutdown. The capacity for such service is not included in the storage capacity of this system.

10.4.15.1.2 System Description

The Nitrogen Supply System is shown on Figure 10.4-18. Nitrogen is supplied to the various components from one or more of the following three types of storage facilities:

1. Bulk Supply

   The following components are supplied by a 2400 psig pressure bulk storage system:

   a. Low Pressure Supply System (100 psig)

      For intermittent use, nitrogen is supplied to the following:

      1. Volume control tank
      2. Pressurizer relief tank
      3. Catalytic recombiners
      4. Spent resin storage tank
      5. Steam generators
      6. Containment Spray System (CSS) chemical additive tank
      7. Steam Generator Blowdown System (SGBS) spent resin storage tank
      8. Waste Processing System (WPS) gas decay tanks
      9. Turbine generator
      10. Vent and Drain System Atmospheric Drain Tanks
      11. Reactor Makeup Water Storage Tank
      12. Boric Acid Storage Tank

   b. High-Pressure Supply System (700 psig)

      For intermittent use, nitrogen is supplied to the Safety Injection System (SIS) accumulators and Pressurizer PORVs (after pressure reduction)

2. Bottle Supply

The Containment Air and Reactor Coolant PASS Remote Operating Modules are supplied with a permanent source of nitrogen via gas cylinders to allow for continuous operation of Post Accident Sampling System when needed.

Nitrogen bottles are also available for the Secondary Sampling Ion Chromatograph and for supporting leak rate testing of the Electrical Penetration Assemblies (EPAs) on an intermittent basis. The bottles are required to be isolated from the EPAs during normal operation to prevent overpressurization.

In addition, nitrogen bottle supply is provided as a backup source to the Plant Gas Nitrogen system in support of the Generator Gas system. A Bottle rack is also located outside the FW Penetration Isolation Valve rooms to provide high pressure nitrogen for valve maintenance purposes.

Nitrogen bottles provide a continuous flow of purge gas to the hydrogen sensor on the RCS hydrogen/oxygen dissolved gas analyzers in the Process Sample System.

3. Temporary Tube Trailer Supply

The following components are supplied from temporary sources which are furnished only when required:

a. Gas decay tanks (initial plant startup)

b. SIS accumulators (initial plant startup)

c. Volume control tank (initial plant shutdown)

d. Pressurizer PORV accumulators (initial plant startup)

The bulk storage supply is common to the two units. Nitrogen is supplied to two high pressure manifolds with each manifold capable of supplying 100 percent of the plant needs. Safety valves are furnished on each manifold as well as downstream of the high-pressure regulators. These valves are set approximately 10 percent above operating pressure. Excessive pressure in the system can be caused by a regulating valve malfunction or a large increase in the stored gas temperature.

Two 700-psig-pressure nitrogen supply lines (one from each manifold) supply the SIS accumulators and the accumulators for the pressurizer PORV operators. Each of the supply lines branches into two more lines, one for Unit 1 and one for Unit 2. In this way, each unit's accumulators are supplied by redundant nitrogen supply manifolds. Pressure and level switches on the accumulators indicate when nitrogen is required. Isolation and feed valves are opened manually from the Control Room.

A second set of pressure regulators in a supply line delivers high-flow 100-psig nitrogen to a header which is common to the two units. This header in turn supplies local headers at each unit. The services supplied by the local headers are grouped in the following manner. Each unit has a separate header which supplies nitrogen to its four steam

generators. Another separate header for the two units supplies nitrogen to the volume control tank, the CSS chemical additive tank, the RCS pressurizer relief tank, steam generator blowdown, and the vent and drain system atmospheric tank. Each unit has a separate line supplied by the main low-pressure header to the turbine generator. Other supply lines from the main header supply the WPS gas decay tanks and hydrogen recombiners, WPS spent resin storage tank, and the SGBS spent resin storage tank. These systems are common to the two units.

Under normal operation, the lines feeding the components which require only intermittent supplies of gas are valved closed, and these feed valves are manually opened only when the need arises, such as during startups, shutdowns, low gas pressure, and so forth. Pressure control is accomplished by pressure regulating valves for each feedline. In addition, each manifold header has a high pressure reducing valve. All regulating valves to distribution points are process actuated and respond to changes in the downstream pressure.

The piping for the Nitrogen Supply System is non-nuclear safety-class, except for those portions which connect to a safety-related component or penetrate the Containment. Where this piping does connect to a safety-related component requiring an isolation valve, this valve and the piping between the valve and the connection are designed to the same safety class as the component. Lines which do not require isolation valves are non-nuclear-safety-class up to the connection of the safety-related component. The nitrogen supply lines which penetrate the Containment are in accordance with ASME B&PV Code, Section III, Safety Class 2, and are discussed in Section 6.2.4.

10.4.15.1.3 Safety Evaluation

The nitrogen supply is a non-nuclear-safety-class system. Safety related components which are intermittently supplied by the system do not require continuous supply during an accident condition. The SIS accumulators, which require nitrogen for safe shutdown of the unit, have sufficient nitrogen within the system to operate as required. Where nitrogen is supplied to a component containing potentially radioactive material, a check valve is provided to prevent contamination of the Nitrogen Supply System.

The bulk storage of the nitrogen is located outdoors where a tank rupture would not affect any safety-related equipment.

10.4.15.1.4 Inspection and Testing Requirements

All piping and pressure vessels for the nitrogen supply system are leak tested in accordance with ANSI B31.1. The systems are inspected periodically to ensure that no leakage exists.

10.4.15.1.5 Instrumentation Applications

The Nitrogen Supply System is furnished with the following instrumentation:

1. Manifold Components (two manifolds per system)

    a. Local pressure indicator

b. Local temperature indicator

2. Supply Header Components (One header per system)

a. Local pressure indicator

b. Low-pressure switch

All low-pressure switches send signals to the local system panel which in turn alarms a single annunciator (“Bulk Gas System N2 Trouble”) in the Control Room to alert the operator to abnormal pressure conditions. Supply lines to the various components are equipped with a pressure indicator (included in the pressure control package) to set and locally monitor gas supply pressure to the component.

Each main and backup supply to the distribution header is furnished with a high-pressure regulating valve which maintains system supply pressure.

Most component supply lines are equipped with a second stage pressure regulator which controls the pressure according to individual component requirements. Table 10.4-16 shows equipment pressures and other pertinent data.

10.4.15.2 Hydrogen Supply System

10.4.15.2.1 Design Basis

The Hydrogen Supply System is designed to provide bulk gas storage capacity to ensure regular replenishment of consumed gases. The system has two manifolds, one for normal operation, the other for standby reserve.

The design storage capacity for the supply package is based on the following:

1. Storage of 30 days is required for gases which are continuously consumed.

2. Storage of one gas charge is required for intermittent uses

3. A safety margin of 10 to 15 percent of Items 1 and 2 is included.

The Hydrogen Supply System is designed to meet the applicable requirements of the ASME B&PV Code, Section VIII, Pressure Vessels, Part A, ANSI B31.1, Code for Pressure Piping, Power Piping, and ANSI B31.8, Gas Transmission and Distribution Piping Systems. This system also meets the applicable National Fire Protection Association (NFPA) codes.

Tables 10.4-16 and 10.4-17 list the components supplied by this system and their requirements. Components requiring only one-time fill prior to startup or shutdown of the unit are not included in the storage capacity of this system.

10.4.15.2.2 System Description

The Hydrogen Supply System is shown on Figure 10.4-19. Hydrogen is supplied by the following systems:

1. Bulk Gas Supply

   The following components are supplied by the bulk gas storage system:

   a. For Intermittent Use

      Gas decay tanks

   b. For Continuous and Intermittent Use

      1. Turbine generator

      2. Primary water tank

      3. Volume Control Tank

2. Bottle Gas Supply

   For continuous use, the following components are supplied by separate bottle gas storage supplies:

   a. Turbine generator and primary water tank (backup to bulk gas storage system)

   b. Reactor coolant drain tank

   The two independent bulk gas storage systems each have redundant supply manifolds. Each supply manifold is capable of supplying 100 percent of the system requirements.

   Safety valves are furnished on each manifold as well as downstream of the high pressure regulators. The safety valves are set approximately 10 percent above operating pressure. Excessive pressure in the system can be caused by regulating valve malfunction or by a large increase in the stored gas temperature.

   The hydrogen bulk gas storage system supplies a header which is common to the two units. This header divides into the NSSS supply line and the turbine generator supply line. The NSSS supply line supplies each unit's volume control tank and the WPS gas decay tanks which are common to the two units. The turbine generator supply line branches into two lines, one for each unit. This line supplies the turbine generator and the primary water tank.

   In addition to the bulk gas supply, the turbine generator and primary water tank for each unit have a backup bottle gas supply. The reactor coolant drain tank for each unit has an independent bottle gas supply. Each of the drain tank supply systems has redundant supply bottles.

Pressure control is accomplished by pressure regulating valves for each feedline. In addition, each manifold header has a high pressure reducing valve. All regulating valves to distribution points are process actuated and respond to changes in the downstream pressure.

Bulk storage tanks are recharged periodically by one or several tube trailers. Truck fill connections are provided on each system for both refilling the supply and furnishing initial charge for components requiring this.

The piping for the Hydrogen Supply System is non-nuclear-safety class except for those portions which connect to a safety-related component or penetrate the Containment. Where this piping does connect to a safety-related component requiring an isolation valve, this valve and the piping between the valve and the connection are designed to the same safety-class as the component. The hydrogen supply lines which penetrate the Containment are in accordance with the ASME B&PV Code, Section III, Safety Class 2, and are discussed in Section 6.2.4.

10.4.15.2.3 Safety Evaluation

The Hydrogen Supply System is a non-nuclear-safety class system. Continuous supply of hydrogen to safety-related equipment is not required for safe shutdown of the plant. Lines to components containing potentially radioactive material are provided with check valves to prevent contamination of the Hydrogen Supply System.

The bulk hydrogen storage is located outdoors where a tank rupture would not affect any safety-related equipment. The outdoor location of these tanks prevents any explosive concentration of hydrogen from accumulating as a result of leaks.

In the event of a break in a hydrogen supply line, the excess flow control valve automatically closes.

10.4.15.2.4 Inspection and Testing Requirements

All piping and pressure vessels for the Hydrogen Supply System are hydrostatically tested to 1.5 times their design pressure or pneumatically tested to at least 1.2 times their design pressure. The systems are inspected periodically to ensure that no leakage exists.

10.4.15.2.5 Instrumentation Application

Each of the normal and backup gas supply systems is furnished with the following instrumentation.

1. Manifold Components (two manifolds per system)

    a. Local pressure indicator

    b. Local temperature indicator

2. Supply Header Components (One header per system)

   a. Local pressure indicator

   b. High- and low-pressure switches

All high- and low-pressure switches send signals to the local system panel which in turn alarms a single annunciator ("Bulk Gas System H2 Trouble") in the Control Room to alert the operator to abnormal pressure conditions. Supply lines to the various components are equipped with a pressure indicator (included in the pressure control package) to set and locally monitor gas supply pressure to the component.

The hydrogen bulk gas storage systems are furnished with high pressure regulating valves on both the main and backup supply manifold. These valves maintain the system supply pressure. Component supply lines are equipped with a second stage pressure regulator which controls the pressure according to individual component requirements.

10.4.16 SECONDARY PLANT SAMPLING SYSTEM

The secondary plant sampling (SPS) system is provided to monitor water chemistry in condensate, steam generator feedwater, steam generator blowdown, main and auxiliary condensers, condensate polishing, heater drains, condensate storage tank, vacuum deaerator, main steam, atmospheric drain tank #1, and the demineralized water storage tank. The main panel is a factory assembled, skid mounted package located at elevation 778 ft in the Electrical & Control Building. Additional remote sample stations are located in the plant and send analyzer signals back to the main panel.

10.4.16.1 Design Basis

The SPS system is designed as the central point to which are routed samples and/or signals from various fluid streams in the plant. A flow diagram of the system is shown on Flow Diagram M1(2)-0222. The pressure and temperature conditions at the various sample points are noted in Table 10.4-18.

Specially designed sample pumps are used to pump the main condenser hot well samples to the sample panel.

The design criteria for the hot well sample pumps, the chilled water return pump, and the sample coolers are given in Table 10.4-19.

10.4.16.2 System Description

The SPS system is designed to continuously measure the selected parameters in samples from the secondary cycle, as shown in Table 10.4-20.

Where required, sample lines are cooled and depressurized by cooling coils, an isothermal bath, and pressure reducing and regulating valves. This process allows for uniform and safe operation of the analyzers and the grab sample sink.

Five remote sample stations are located outside the main panel room and transmit analyzer signals back to the main panel.

1. Located in the Safeguards Building pipe tunnel, the Condensate Storage Tank upper and lower volumes are sampled for dissolved oxygen, cation conductivity, and sodium ion (local panel only).

2. Located in the Turbine Building (adjacent to the condensate pumps), the Atmospheric Drain Tank #1 is sampled for cation conductivity.

3. Located in the Turbine Building (adjacent to the vacuum deaerator), the Vacuum Deaerator is sampled for dissolved oxygen and the Demineralized Water Storage Tank is sampled for specific conductivity.

4. Located in the Turbine Building are two analyzers which monitor and display condenser sodium concentration from hotwells A and B. Signals are sent to the main sampling panel recorders. High sodium concentration alarm signals are announced on the main sampling panel.

A Secondary Side Ion Chomatograph is located in its own enclosure in the turbine building (near the circulating water booster pumps). The analyzer, common to both units, takes samples of Unit 1 and 2 hotwell A.

Local Dissolved Oxygen Monitors are located at various locations in the Turbine Building including one at the Feedwater sampling station on elevation 803' and at Common Condensate Discharge Monitoring station on elevation 778'. Data is relayed to the Plant Computer.

Secondary Corrosion Product Monitor panels are provided at Feedwater, Steam Generator Blowdown, Heater Drains and Condensate sampling stations located throughout the Turbine Building and Electrical Control Building.

10.4.16.3 Safety Evaluation

This system is non-nuclear-safety-related. Seismic design criteria are not applicable. Safe shutdown of the unit does not rely on the SPS system. Hence, there is no provision to provide power to this system from emergency diesel generators.

10.4.16.4 Tests and Inspections

All tests and inspections of the SPS system are performed in accordance with applicable codes.

10.4.16.5 Instrumentation and Control Requirements

It is intended that the SPS system perform only advisory and early warning functions (except for amine and hydrazine pump control) by continuously monitoring critical parameters in certain fluid streams; therefore, an extensive system alarm annunciator is provided. Activation of any of the alarms is automatically relayed to the common trouble alarm located on the main control board.

REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 60, Control of Releases of Radioactive Material to the Environment.

2. 10 CFR Part 50, Appendix A, General Design Criterion 64, Monitoring Radioactivity Releases.

3. NRC Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam- and Radioactive-Waste-Containing Components of Nuclear Power Plants, Revision 3, February 1976.

4. ANSI N18.2, Nuclear Safety Criteria for the Design of Pressurized Water Reactors.

5. NRC Regulatory Guide 1.29, Seismic Design Classification, Revision 2, February 1976, U.S. Nuclear Regulatory Commission.

6. Branch Technical Position APCSB 3-1, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, attached to Standard Review Plan 3.6.1.

7. Branch Technical Position MEB 3-1, Postulated Break and Leakage Locations in Fluid System Piping Outside Containment, attached to Standard Review Plan 3.6.2.

8. 10 CFR Part 50, Appendix A, GDC 2, Design Bases for Protection Against Natural Phenomena.

9. 10 CFR Part 50, Appendix A, GDC 4, Environmental and Missile Design Bases.

10. 10 CFR Part 50, Appendix A, GDC 5, Sharing of Structures, Systems, and Components.

11. 10 CFR Part 50, Appendix A, GDC 19, Control Room.

12. 10 CFR Part 50, Appendix A, GDC 44, Cooling Water.

13. 10 CFR Part 50, Appendix A, GDC 45, Inspection of Cooling Water System.

14. 10 CFR Part 50, Appendix A, GDC 46, Testing of Cooling Water System.

15. Branch Technical Position APCSB 10-1, Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants.

16. NRC Regulatory Guide 1.70, Standard Format & Content of Safety Analysis Report for Nuclear Power Plants, LWR edition, Revision 2, September 1975.

17. 10 CFR Part 50, Appendix A, GDC 57, Closed System Isolation Valves.

18. NUREG-0291, An Evaluation of PWR Steam Generator Water Hammer by Creare, Inc, Dec. 31, 1976.

19. WCAP-09364, Vol. 1, High Pressure Water Hammer Test Program for the Counterflow Preheat Steam Generator.

20. Westinghouse Report “Counterflow Preheat Steam Generator Tube Expansion Report,” June 1983.

21. Westinghouse Report “Counterflow Preheat Steam Generator Vibration Summary,” June 1983.

22. NRC Regulatory Guide 1.32, “Criteria for Safety Related Electric Power Systems for Nuclear Power Plants”, Revision 2, February 1977, U. S. Nuclear Regulatory Commission.

23. NRC Regulatory Guide 1.47, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems, May 1973, U. S. Nuclear Regulatory Commission.

24. NRC Regulatory Guide 1.53, Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems, June 1973, U. S. Nuclear Regulatory Commission.

25. NRC Regulatory Guide 1.75, Physical Independence of Electric Systems, Revision 1, January 1975, U. S. Nuclear Regulatory Commission.

26. Westinghouse Guidelines for Secondary Water Chemistry (SGT-5.1.1-4468), dated February 1985.

27. EPRI PWR Secondary Water Chemistry Guidelines, Revision 2, dated November 1988.

28. 10CFR50, Appendix A, GDC 54, Piping Systems Penetrating Containment.

TABLE 10.4-1
CONDENSER EVACUATION SYSTEM EQUIPMENT DESIGN CRITERIA

Condenser Exhausting Vacuum Pump Package Data Sheets

a. Condenser Exhausting Vacuum Pump

Type of Pump	Rotary, two stage
Design holding capacity, scfm at 1.0 in. Hg abs.	25
Horsepower input to pump at design conditions, bhp	130
Speed, rpm	
Pump	435
Motor	1800
Motor rated hp/service factor	150/1.15
Motor, voltage/hertz/phase	460/60/3
Hogging capacity, dry air, acfm at 10 in. Hg. abs.	2300
Maximum flow rate of air, vapor at discharge, scfm at 15 in. Hg abs.	1250

b. Condenser Exhausting Vacuum Pump Heat Exchanger

Shell-side design temperatures, °F	300
Shell-side design flow rate, gpm	90
Tube-side design temperatures, °F	300
Tube-side design velocity, ft/sec	5.3
Tube-side design flow rate, gpm	700
Shell-side design pressure, psig	150
Tube-side design pressure, psig	75
Tube-side pressure drop (ft)	5.075

TABLE 10.4-2
STEAM DUMP VALVES DESIGN DATA

Safety class	Non-nuclear-safety-related, non-seismic Category I
Design pressure, psia	1300
Design temperature, °F	700
Operating steam flow, lb/hr/valve	530,000
Operating pressure, psia	965 to 1107
Operating temperature, °F	540 to 557
Maximum steam flow, lb/hr/valve (at 1185 psig pressure)	1,109,000
Valve opening, sec	3
Valve closing, sec	5
Actuator	Pneumatic
Size, in.	8

TABLE 10.4-3
ANTICIPATED WATER QUALITY OF SQUAW CREEK RESERVOIR

Substance	Range as CaCo3 (mg/l)
Calcium	360 to 940
Magnesium	80 to 410
Sodium	1810 to 2220
Bicarbonate	200 to 210
Carbonate	0 to 28
Sulfate	620 to 1040
Chloride	1630 to 2310
Silica	8 to 68 (as Si02)
Ammonia	1.2 (as N2)
Phenolphthalein alkalinity	0 to 8
Methyl orange alkalinity	200 to 238
Hardness	400 to 1350 ppm
Total dissolved solids	2450 to 3570 ppm

TABLE 10.4-4
(DELETED)

TABLE 10.4-5
CONDENSATE AND FEEDWATER SYSTEM EQUIPMENT DESIGN PARAMETERS

Steam Generator Feedwater Pumps	
Safety class	NNS
Capacity at design rating, gpm	19,800
Capacity at normal operation, gpm (356°F)	17,090
TDH at design rating, ft	2322
TDH at normal operation, ft	2278
Minimum NPSH required at design rating, ft	232
Pump efficiency at design rating, percent	88.6
Design pressure, psig	2000
Design temperature, °F	400
Condensate Pumps	
Safety class	NSS
Capacity at design rating, gpm	11,460
Capacity at normal operation, gpm (121°F)	9950
TDH at design rating, ft	1095
TDH at normal operation, ft	1120
Minimum recirculation flow, gpm	2900
TDH at minimum recirculation flow, ft	1340
Condensate Storage Tank	
See Table 10.4-8 and Section 10.4.9	

Note:

Total dynamic head (TDH); net positive suction head (NPSH)

TABLE 10.4-6
STEAM GENERATOR BLOWDOWN SYSTEM EQUIPMENT DESIGN CRITERIA

(Sheet 1 of 4)

SGBS Heat Exchanger	
Quantity	One
Type	Horizontal U-tube
TEMA class	R
Tag Nos.	CP1-SBAHSB-01 and CP2-SBAHSB-01
Operating mode	1 x 100 percent
Design heat transfer rate, Btu/hr	150.73×10^6
Effective heat transfer area, ft^2	15,200
Overall length, in.	362
Outside diameter, in.	84

	Tube Side	Shell Side
System fluid	SGB	Condensate and SGB
Design flow, lb/hr	330,000	635,000
Design pressure, psig	1200	300
Design temperature, °F	700	600
Inlet temperature, °F	567	125
Outlet temperature, °F	130	361
Thermal relief valves	One	One
Set pressure, psig	1200	300
Material	SA 249, Type 304	SA 516, Grade 70

TABLE 10.4-6
STEAM GENERATOR BLOWDOWN SYSTEM EQUIPMENT DESIGN CRITERIA

(Sheet 2 of 4)

SGBS Inlet Filters	
Quantity	Two
Tag Nos.	CP1-SBFLSB-01 and CP1-SBFLSB-02
Design temperature, °F	150
Design pressure, psig	300
Design flow, gpm	360
Dirt-holding capacity, grams of ferric oxide	1320
Material	AISI 304 stainless steel
Micron Rating (μM)	0.45 to 40 absolute (100% retention)
SGBS Spent Resin Sluice Pump Filter	
Quantity	One
Tag No.	CP1-SBFLSR-01
Design Temperature, °F	150
Design pressure, psig	150
Design flow, gpm	140
Dirt-holding capacity, grams of ferric oxide (Fe304)	3204
Material	AISI 304 stainless steel
Micron Rating (μM)	0.45 to 100 absolute (100% retention)

TABLE 10.4-6
STEAM GENERATOR BLOWDOWN SYSTEM EQUIPMENT DESIGN CRITERIA

(Sheet 3 of 4)

Steam Generator Blowdown System Demineralizers

Quantity	Four
Tag Nos.	CP1-SBDMCD-01 CP1-SBDMCD-02 CP2-SBDMAD-01 CP2-SBDMAD-02
Type	Flushable
Design pressure, psig	300
Design temperature, °F	150
Design flow, gpm	640
Resin type	Commercial available resins determined to be acceptable for use in the Steam Generator Blowdown system by the Secondary Chemistry Control Program.
Resin volume, ft^3	80
Overall Dimensions	
Outside diameter, in.	60
Shell height, in.	96
Material	AISI 304 stainless steel
Decontamination factor	See Table 11.2-6.

TABLE 10.4-6
STEAM GENERATOR BLOWDOWN SYSTEM EQUIPMENT DESIGN CRITERIA

(Sheet 4 of 4)

Steam Generator Blowdown Spent Resin Sluice Pump

Quantity	One
Tag No.	CPX-SBAPRS-01
Operating mode	1 x 100 percent
Type	Horizontal centrifugal, canned
Designed flow, gpm	150
Design heat, ft	230
Material	AISI 316 stainless steel
Motor	
hp	20
Service factor	1.15
rpm	3450
V/phase/Hz	460/three/60

Steam Generator Blowdown Spent Resin Storage Tank

Quantity	One
Tag No.	CPX-SBATSR-01
Safety class	NNS
Type	Vertical cylindrical
Design capacity, gal	3740
Design pressure, psig	100
Design temperature, °F	200
Material	AISI 304 stainless steel

TABLE 10.4-7
STEAM GENERATOR BLOWDOWN CLEANUP SYSTEM FAILURE ANALYSIS

Components	Malfunction	Remarks
SGBS isolation valves	One valve fails closed.	Blowdown from one steam generator is not available. Plant safety is not affected.
	One valve fails open	These valves are designed to fail closed. Manual valves are provided in series with the automatic valves.
Isolation valves downstream of the SGBS heat exchanger	One valve fails closed. One valve fails open.	Plant safety is not affected.
Demineralizers (cation or mixed bed or both)	One demineralizer fails or is exhausted.	Decontamination capability is reduced temporarily. Plant safety is not affected.

TABLE 10.4-8
EQUIPMENT DESIGN PARAMETERS

(Sheet 1 of 3)

1. Motor-Driven Auxiliary Feedwater Pump

Safety class	3
Seismic class	Category I
Design flow, gpm	570
Design TDH, ft	3160
Shutoff TDH	3650
Normal flow	N/A
Normal TDH	N/A
Casing design temperature, °F and pressure, psig	150/1600
Runout flow, gpm	800
Minimum flow, gpm	100
Efficiency at design point, percent	73
Design point bhp	645
Maximum bhp	690
Motor hp and service factor	700/1.15
Motor electrical requirements, V/Hz/phase	6600/60/3
Rpm	3560
Miscellaneous information	horizontal, split-casing, centrifugal pump, Ingersoll Rand pump type 4HMTA-9 stage
Material, casing	ASME SA-216, WCB
Impeller material	ASME A-296-CA6NM
Suction and discharge nozzle material	ASME SA-216, WCB

TABLE 10.4-8
EQUIPMENT DESIGN PARAMETERS

(Sheet 2 of 3)

2. Turbine-Driven Auxiliary Feedwater Pump	
Safety class	3
Seismic class	Category I
Design flow, gpm	1145
Design TDH, ft	3160
Shutoff TDH	3722
Normal flow	N/A
Normal TDH	N/A
Casing design temperature, °F, and pressure, psig	150/1700
Runout flow, gpm	2118
Minimum flow, gpm	100
Efficiency at design point, percent	76.5
Design point bhp	1194
Maximum bhp	1300
Rpm	4075
Miscellaneous information	horizontal, split-casing, centrifugal pump, Ingersoll Rand pump type 5HMTA-6 stage
Material, casing	ASME SA-216, WCB
Impeller material	ASME A-296-CA6NM
Suction and discharge nozzle material	ASME SA-216, WCB
3. Condensate Storage Tank	
Safety class	3
Seismic class	Category I
Design pressure, psig and temperature, °F	0/150

TABLE 10.4-8
EQUIPMENT DESIGN PARAMETERS

(Sheet 3 of 3)

Total Volume of Tank, gal	500,000
Medium	condensate
Normal operating pressure and temperature, °F	atmospheric/40-120
Material of construction	concrete with stainless steel liner
Type of vessel and orientation	cylindrical/vertical
Diaphragm	Yes
Atmospheric relief	Yes
Miscellaneous	vertical, concrete, site-fabricated, outdoor tank

TABLE 10.4-9
FAILURE MODE ANALYSIS

(Sheet 1 of 7)

BREAK ON FEEDWATER PIPE TO STEAM GENERATOR NO. 4 [NOTE 1], INSIDE CONTAINMENT AND SINGLE FAILURE

Failure	Equipment [Note 2]	Function	Failure Mode	Effect of Failure	Analysis
1	Diesel Generator A (active failure)	Provides power to Auxiliary feedwaterr pump motor 1 and motor controlled valve 17	Diesel Generator A fails to start		Meets single failure criteria:
				a. Auxiliary feedwater pump 1 is inoperative.	a. AF Pumps 2 and 3 are operable
				b. Steam generators 1 and 2 cannot be supplied by pump 1.	b. Steam generators 1 and 2 supplied from AF Pump 3 (Turbine Driven). Steam generator 3 supplied from AF Pumps 2 and 3.
				c. Valve 17, Train A, cannot be closed from Control Room	c. Valves 12 and 13 are closed from Control Room manually. (Isolates SG 4 Break).
2	Diesel Generator B (active failure)	Provides power to Auxiliary feedwater pump motor 2 and motor controlled valve 16	Diesel Generator B fails to start		Meets single failure criteria:
				a. Auxiliary feedwater pump 2 is inoperative.	a. AF Pumps 1 and 3 are operable.
				b. Steam generators 3 and 4 cannot be supplied by pump 2.	b. Steam generators 1 and 2 supplied from AF Pumps 1 and 3 and Steam Generator 3 supplied from AF Pump 3.
				c. Valve 16, on Train B, cannot be closed from Control Room.	c. Valves 12 and 13 are closed from Control Room manually. (Isolates SG 4 Break).

TABLE 10.4-9
FAILURE MODE ANALYSIS

(Sheet 2 of 7)

BREAK ON FEEDWATER PIPE TO STEAM GENERATOR NO. 4 [NOTE 1], INSIDE CONTAINMENT AND SINGLE FAILURE

Failure	Equipment [Note 2]	Function	Failure Mode	Effect of Failure	Analysis
3	Battery A System (passive failure)		Battery A System lost		Meets single failure criteria:
		a. Pump 1 motor circuit breaker		a. Auxiliary feedwater pump 1 is inoperative.	a. AF Pumps 2 and 3 are operable.
		b. Provides control power to valves 6, 8, 11 and 13.		b. Steam generators 1 and 2 cannot be supplied by pump 1.	b. Steam generators 1 and 2 are supplied from AF Pump 3. Steam Generator 3 supplied from AF Pumps 2 and 3.
		c. AF Pump 3 speed indication		c. Valves 6, 8, 11 and 13 fail open.	c. Valves 12 and 16 are closed from Control Room manually. Throttling control of valves 6, 8, 11 and 13 is not available in Control Room; requires manual action locally.
		d. Provides control power to recirc. Valve 14		d. No remote speed indication on AF Pump 3.	d. AF Pump 3 runs at highest speed of speed setter. Pump may be tripped locally if required.
				e. AF Pump 1 recirc. valve open.	e. AF Pump 1 already inoperative, no effect.
3A	118 Vac “A” instrumentation system (passive failure)	a. Speed control for AF Pump 3	118 Vac “A” Bus Lost	Signal from speed set lost.	Same as 3d above.

TABLE 10.4-9
FAILURE MODE ANALYSIS

(Sheet 3 of 7)

BREAK ON FEEDWATER PIPE TO STEAM GENERATOR NO. 4 [NOTE 1], INSIDE CONTAINMENT AND SINGLE FAILURE

Failure	Equipment [Note 2]	Function	Failure Mode	Effect of Failure	Analysis
4	Battery B System (passive failure)	a. Pump 2 motor circuit breaker b. Provides control power to valves 7, 9, 10 and 12 c. Provides control power to recirc. valve 15.	Battery B System lost	a. Auxiliary feedwater pump 2 is inoperative. b. Steam generator 3 cannot be supplied by pump 2. c. Valves 7, 9, 10 and 12 fail open. d. AF Pump 2 recirc. valve opens.	Meets single failure criteria: a. AF Pumps 1 and 3 are operable. b. Steam generators 1 and 2 are supplied from AF Pump 1 and 3. Steam generator 3 is supplied from AF Pump 3. c. Valves 13 and 17 closed manually from Control Room. Throttling control of valves 7, 9, 10 and 12 is not available in Control Room; requires manual action locally. d. AF Pump 2 inoperative, no effect.
5	Train A cable system (passive failure)	See failures 1, 3, and 3A.	See failures 1, 3 and 3A	See failures 3 and 3A	Meets single failure criteria: See failures 3 and 3A.
6	Train B cable system (passive failure)	See failures 2 and 4.	See failures 2 and 4.	See failure 4.	Meets single failure criteria: See failure 4.

TABLE 10.4-9
FAILURE MODE ANALYSIS

(Sheet 4 of 7)

BREAK ON FEEDWATER PIPE TO STEAM GENERATOR NO. 4 [NOTE 1], INSIDE CONTAINMENT AND SINGLE FAILURE

Failure	Equipment [Note 2]	Function	Failure Mode	Effect of Failure	Analysis
7	Motor driven pump No 1	Supplies water to Steam Generators 1 and 2.	Stops pumping.	Steam Generators 1 and 2 cannot be supplied by Pump 1.	Meets single failure criteria: AF Pumps 2 and 3 are operable. 1) Motor-driven pump No. 2 feeds steam generator No. 3 while valve 12 (or 17) is manually closed from Control Room. 2) Turbine driven pump No. 3 feeds Steam Generators No. 1, 2 and 3 while valve 13 (or 16) is manually closed from Control Room.
8	Turbine driven pump (No. 3)	Supplies water to Steam Generators 1, 2, 3 and 4.	Does not operate. Loss of Turbine Lube Oil.	Steam Generators 1, 2, 3 and 4 cannot be supplied by Pump 3.	Meets single failure criteria: AF Pumps 1 and 2 are operable. 1) Motor-driven pump No. 1 feeds Steam Generators No. 1 and 2. 2) Motor-driven pump No. 2 feeds Steam Generator No. 3 while valve 12 (or 17) and 13 (or 16) is manually closed from Control Room.
9	Valve No. 12	Isolates Pump No. 1 flow to Steam Generator No. 4.	Fails to close	Valve No. 12 cannot isolate faulted steam generator.	Meets single failure criteria: Valve 17 closed manually from Control Room.
10	Valve No. 13	Isolates Pump No. 3 flow to Steam Generator No. 4.	Fails to close	Valve No. 13 cannot isolate faulted steam generator.	Meets single failure criteria: Valve 16 closed manually from Control Room.

TABLE 10.4-9
FAILURE MODE ANALYSIS

(Sheet 5 of 7)

BREAK ON FEEDWATER PIPE TO STEAM GENERATOR NO. 4 [NOTE 1], INSIDE CONTAINMENT AND SINGLE FAILURE

Failure	Equipment [Note 2]	Function	Failure Mode	Effect of Failure	Analysis
11	Motor Control	Motor operation.	a) Loss of control power b) Any control circuit fault	Motor will not start	Meets single failure criteria: Similar to Failures 1, 2, 3, 3A,5, 6, and 7
12	Turbine Control	Turbine speed control.	a) Loss of control power b) Loss of instrument air	Speed control goes to rated hi-speed control.	Meets single failure criteria: Steam admission valves fail open on loss of elec. or air (See Failure Nos. 3 and 3A) Turbine speed control goes to hi-speed on loss of elec or air. Shut valves 16 and 17 remotely.
13	AO Feed Regulator Valves (6, 8, 10, 12) (Motor-driven pumps)	Regulate Flow to Steam Generators/Isolate faulted Steam Generator.	a) Loss of air b) Loss of control power c) "Trip to Auto" fails	a) No effect for 30 minutes, the valves fail open. b) On loss of signal valves fails open c) Valve remains in manual control	Meets single failure criteria: a) Air accumulator ensure isolation function for 30 minutes and allows up to 30 minutes of flow control after which operator controls valves locally. Shut valves 16 and 17 remotely b) Valve controlled manually or pump shutdown (See Failure Nos. 3 and 3A). Shut valves 16 & 17 remotely. c) Same as b) above.

TABLE 10.4-9
FAILURE MODE ANALYSIS

(Sheet 6 of 7)

BREAK ON FEEDWATER PIPE TO STEAM GENERATOR NO. 4 [NOTE 1], INSIDE CONTAINMENT AND SINGLE FAILURE

Failure	Equipment [Note 2]	Function	Failure Mode	Effect of Failure	Analysis
14	Pump recirc. control	Isolates miniflow to ensure required flow to Steam Generators.			Meets single failure criteria:
			a) Loss of air	a) No effect for 30 minutes, then valve fails open.	a) Air accumulator ensures 30 minutes of isolation after which operator controls recirculation manually.
			b) Loss of control power or signal	b) Recirc. valve fails open	b) See Failure 3 and 3A.
15	AO Feed Regulator Valves (7, 9, 11, 13) (Turbine-driven pump)	Regulate flow to Steam Generator/Isolate faulted Steam Generator.			Meets single failure criteria:
			a) Loss of air	a) Accumulator supplies 30 minutes of air to allow remote isolation of a faulted steam generator, then valve fails open	a) Remote manual control available to isolate AF flow for 30 minutes. Then resort to local manual manipulation of valves. AF flow assured as all valves fail open.
			b) Loss of control power	b) Remote manual capability lost..	b) Flow control is not required for 30 minutes. Shut valves 16 and 17 remotely.
16	Train A SSPS Fails	Starts Aux Feedwater Pump 1 and opens Steam Admission Valve on MSL 4	Loss of Inverter		Meets single failure criteria:
				a. Aux Feed Pump 1 does not start.	a. Steam generators 1 and 2 supplies from AF Pump 3 (Turbine Driven). Steam Generator 3 supplied from AF Pump 2 and 3.
				b. Steam Admission Valve on MSL 4 does not open.	b. Steam Admission Valve on MSL Loop 1 opened by Train B SSPS.

TABLE 10.4-9
FAILURE MODE ANALYSIS

(Sheet 7 of 7)

BREAK ON FEEDWATER PIPE TO STEAM GENERATOR NO. 4 [NOTE 1], INSIDE CONTAINMENT AND SINGLE FAILURE

Failure	Equipment [Note 2]	Function	Failure Mode	Effect of Failure	Analysis
17	Train B SSPS Fails	Starts Aux Feedwater Pump 2 and opens Steam Admission Valve on MSL 1	Loss of Inverter		Meets single failure criteria
				a. Aux Feedwater Pump 2 does not start.	a. Steam Generators 1 & 2 are fed by AF Pump 1
				b. Steam Admission Valve on MSL 1 does not open.	b. Steam Admission Valve on MSL 1 is manually opened.

Notes:

1 Pipe break assumed on line to steam generator 4 for failure mode analysis. Analysis and results for breaks on lines to steam generators 1, 2 or 3 are similar.

2 The equipment numbers are also shown on Figure 10.4-12.

TABLE 10.4-10
MAXIMUM SYSTEM HEAT SINK REQUIREMENTS

(Sheet 1 of 3)

Component To Be Cooled	Number Provided	Number in Operation	Total Cooling Water Flow Rate (gpm)	Total Operating Heat Load (Btu/hr)
Steam generator feedwater pump turbine drive oil coolers	4	2	480	$.99 \times 10^6$
Electrohydraulic control fluid	2	1	200	0.496×10^6 coolers
Turbine lube oil coolers	3	2	3700	14.77×10^6
Heater drain pump lube oil	2	2	35	0.005×10^6 coolers
Condensate pump motor	2	2	34	0.612×10^6 bearing coolers
Exciter air coolers	2	2	792.5	2.184×10^6
Hydrogen coolers	8	8	4403	18.532×10^6
Isophase forced cooling unit	2	1	220	1.91×10^6
Primary water coolers	6	4	6000	32.696×10^6
Primary water leakage unit	1	1	154	0.83×10^6

TABLE 10.4-10
MAXIMUM SYSTEM HEAT SINK REQUIREMENTS

(Sheet 2 of 3)

Component To Be Cooled	Number Provided	Number in Operation	Total Cooling Water Flow Rate (gpm)	Total Operating Heat Load (Btu/hr)
Generator seal oil unit	2	1	66	0.45×10^6
Rotary-Screw Instrument air compressor CPX-CICACO-01	1	1 (common)	40	0.46×10^6
Rotary-Screw Instrument air compressor package CPX-CICACO-02	1	1 (common)	40	0.46×10^6
Auxiliary steam condensate cooler	1	1 (common)	150	4.65×10^6
Auxiliary steam condensate sample cooler	1	1 (common)	5	0.025×10^6
Secondary sampling system	1	1	50	1.25×10^6
Auxiliary boiler sample cooler and conductivity cell	1	1 (common)	1	$.0225 \times 10^6$
Condensate polishing system Vacuum deaerator vacuum pump	1	1	21	0.158×10^6
seal water coolers	2	2	72	0.14×10^6

TABLE 10.4-10
MAXIMUM SYSTEM HEAT SINK REQUIREMENTS

(Sheet 3 of 3)

Component To Be Cooled	Number Provided	Number in Operation	Total Cooling Water Flow Rate (gpm)	Total Operating Heat Load (Btu/hr)
Generator H_2 Gas Dryer Regen. Cycle Cooler (Thermer)	1	1	1	$.0051 \times 10^6$
TOTAL	46	38	16,194	79.9266×10^6

TABLE 10.4-11
ANTICIPATED ANALYSIS OF THE TURBINE PLANT COOLING WATER

1.	Corrosion Inhibitor A suitable corrosion inhibitor is used.	
2.	pH at 25°C	> 8.0
3.	Chloride, mg/l (max.)	10.0

TABLE 10.4-12
ANTICIPATED ANALYSIS OF THE CIRCULATING WATER

Substance	mg/l, as Calcium Carbonate (except as noted)
Calcium	560 to 940
Magnesium	80 to 410
Sodium	1810 to 2220
Bicarbonate	200 to 210
Carbonate	0 to 28
Sulfate	620 to 1040
Chloride	1630 to 2310
Silica, as silicon dioxide	8 to 68
Ammonia, as N	1 to 2
Phenolphthalein alkalinity	0 to 8
Methyl orange alkalinity	200 to 238
Total hardness	640 to 1350
Total dissolved solids	2450 to 3570
pH at 25 °C	8.0 to 8.2
Turbidity as Jackson Units	5 to 10

Note:

The composition of this water is based on predicted average values for the SCR. The peak values may be somewhat higher than the listed values.

TABLE 10.4-13
AUXILIARY STEAM SYSTEM REQUIREMENTS

Auxiliary Steam Requirement	Point Required	Maximum Flow (lb/hr)	Steam Conditions (psig)
Chemical Pipe cleaning	Prior to startup	As required	50 to 150 psig SAT
Turbine Components			
Turbine Gland Steam System			
a. Main Turbine	During Startup	6,832(Normal) 11,614(Enlarged)	60-150 psig SAT.
b. FWP Turbine	During Startup	1,150(575 Each)	60-150 psig SAT.
	Power Operations	230 (115 Each)	60 - 150 psig SAT.
Main Steam Reheat Steam Dump System (Reheater Tube Sheet Warmup)	During Startup (After Hotwell Sparger Warmup)	As required	150 psig SAT.
Condensate System (Hot Well Sparger)	During Startup	10,000	150 psig SAT.
NSSS Components			
Floor Drain Waste Evaporator Package	During Evaporator Operation	10,600	50 psig SAT.
WPS Waste Evaporator Package	During Evaporator Operation	10,600	50 psig. SAT.
BRS Recycle evaporator package	During Evaporator Operation	10,600	50 psig SAT.
CVCS Boric Acid Batching Tank	During Weekly Batches	500	50 psig SAT.

TABLE 10.4-14
AUXILIARY STEAM SYSTEM REQUIREMENTS - NSSS

Auxiliary Steam Requirement	Steam Flow at 50 psig and 297 °F (lb/hr)	Condensate Return Pressure, Minimum to Maximum (psig)
Evaporator Packages (Each 15 gpm) Floor Drain Waste, WPS Waste, BRS Recycle		
Feed preheater	1400	5 to 15
Evaporator	9100	5 to 45
Evaporator condenser	100	none
Boric acid batching tank heater	500	40 (maximum)

TABLE 10.4-15
HAS BEEN DELETED

TABLE 10.4-16
NITROGEN AND HYDROGEN SUPPLY REQUIREMENTS

(Sheet 1 of 3)

Component	Gas	Operating Pressure (psig)	Operating Temperature (°F)	Flowrate (Normal/Max.)	Service (Continuous or Intermittent)	Normal Consumption	Comments
Volume control	1. Hydrogen	15-60	115	a. 0.7 scfm	Continuous Intermittent	1000 scf per day	
				b. 30 scfm		1000 scf per startup	
	2. Nitrogen	10-15	Ambient	Variable	Required for initial shutdown	4500 scf	Fills one shutdown gas decay tank which is recycled for subsequent shutdown
Reactor Coolant (per tank)	Hydrogen	2-6	120-170	Negligible	Continuous	Normally none	Provides H2 makeup; drain tank processed H2 is mainly from reactor coolant. Separate H2 supply to be provided (388 scf)
Pressurizer relief tank (per tank)	Nitrogen	0-3	Ambient	a. Negligible	Intermittent	Negligible	Cover gas (not recycled) N2 to permit pressurizer draining during shutdown (not recycled)
				b. Variable	Intermittent	2000 scf	
Gas decay tanks (8 tanks)	Nitrogen	4	Ambient	Variable	Required for initial plant startup	6500 scf	Used to initially inert and to pressurize gas decay tanks to provide sufficient suction for waste gas compressors

TABLE 10.4-16
NITROGEN AND HYDROGEN SUPPLY REQUIREMENTS

(Sheet 2 of 3)

Component	Gas	Operating Pressure (psig)	Operating Temperature (°F)	Flowrate (Normal/Max.)	Service (Continuous or Intermittent)	Normal Consumption	Comments
Catalytic recombiner (per recombiner)	Nitrogen	0	Ambient	Variable	Intermittent	By user	Used for initial inerting and recombiner) maintenance
Spent resin storage tank	Nitrogen	a. 0.5-2.0	Ambient	Negligible	Continuous	Negligible	Cover gas (not recycled)
		b. 90 max.	Ambient	Variable	Intermittent	1200 scf	Provides motive force to discharge spent resin to bulk disposal (not recycled)
SIS accumulators (per tank)	Nitrogen	660	Ambient	a. Variable	Continuous	Variable	Usage dependent on containment temperature variations (not recycled)
				b. Variable	Intermittent	22000 scf	Required for initial pressurization
Steam generators (per steam generator)	Nitrogen	2	Ambient	Variable	Intermittent	6000 scf dry layup 1500 scf wet layup	Inerting
CSS chemical additive tank (per tank)	Nitrogen	2	Ambient	Negligible	Intermittent	134 scf	For cover gas
SGBD spent resin storage tank	Nitrogen	a. 0.5-2.0	Ambient	Negligible	Continuous	Variable	Cover gas
		b. 90 max.	Ambient	Variable	Intermittent	1200 scf	To force spent resin to disposal

TABLE 10.4-16
NITROGEN AND HYDROGEN SUPPLY REQUIREMENTS

(Sheet 3 of 3)

Component	Gas	Operating Pressure (psig)	Operating Temperature (°F)	Flowrate (Normal/Max.)	Service (Continuous or Intermittent)	Normal Consumption	Comments
Boron Recycle System Evaporator Condenser	Nitrogen	1-5	250	15-25 scfm	Intermittent	Variable	To provide additional NPSH for the concent rator pump during pump-out
Vent and Drain Atmospheric Drain Tanks (per tank)	Nitrogen	<1 psig	Ambient	1 scfm	Continuous	1 scfm	Required during Heater Drain Pump Operation
Secondary Sampling Ion Chromatograph	Nitrogen	90-100	Ambient				
Reactor Makeup Water Storage Tank	Nitrogen	<1	Ambient	≤40 scfh	Intermittent or Continuous	Variable	For sparging. Nitrogen is supplied into RMWP Recirculation Line
Reactor Makeup Water Storage Tank	Nitrogen	<1/3 psig	Ambient	Negligible	Continuous	Variable	For Blanketing
Demineralized water storage tank DWST Sparge/Purge	Nitrogen	30	Ambient	Variable	Continuous	0-30 scfm	Sparge/Purge gas
		80	Ambient	Variable	Continuous	0-300 scfm	
PASS Purge System	Nitrogen	1800	Ambient		Intermittent		Required for Post Accident Sampling to flush sample lines.
PS RCS H2/O2 Analyzer	Nitrogen	75	Ambient	0.011 scfm	Continuous	0.011 scfm	Purge gas to H2 sensor
Boric Acid Storage Tank	Nitrogen	100	Ambient	≤50 scfh	Intermittent	Variable	Sparge gas

TABLE 10.4-17
TURBINE GENERATOR SYSTEM

Component	Gas	Operating Pressure (psig)	Operating Temperature (°F)	Service Pressure	Service (Continuous or Intermittent)	Normal Consumption	Comments
Nitrogen supply	N2	Bottle pressure, 2844 psig; downstream of pressure reducer, 5.7-21.5 psig	Room temperature	0.2-0.3 bar[(a)] 2.8-4.3 psig	Only on draining and priming of primary water unit	With Allis-Chalmers units. One bottle of a volume of 10Nm$^{3(a)}$ (= 353 ft^3) is provided.	Purging of primary water tank
Nitrogen supply	N2	Bottle pressure, 2844 psig; downstream of pressure reducer, 5.7-21.5 psig	Room temperature	0.428 psig	Only on defect of measuring device	With Allis-Chalmers units. One bottle of a volume of 10Nm$^{3(a)}$ (= 353 ft^3) is provided.	
Hydrogen supply	H2	200-300 psig of central supply	Room temperature	Approximately 3 psig	Continuous	Approximately 200 Nl/h[(a)] =7.06 scf/h	Gas cushion over primary water tank
Hydrogen supply	H2	200-300 psig of central supply	Room temperature	14-70 psig	Only on priming	2.5 x generator volume + 1 x generator volume per 14.7 psi pressure increase	Priming of generator
Hydrogen supply	H2	200-300 psig of central supply	Room temperature	5.1 bar[(a)] 75 psig	Continuous	Losses 500 Nl/h[(a)] =424 scf/day Purity Measurement 450 nl/day =18.75 nl/h[(a)] =0.66 scf/h	Maintaining of H2 concentration in generator casing

Notes:

a) Metric units:

Nl/h = Normal liters per hour (at atmospheric pressure and standard temperature)

Nm^3 = Normal cubic meters (at atmospheric pressure and standard temperature)

bar = 14.7 psia (1 atmosphere)

TABLE 10.4-18
SAMPLE POINTS PRESSURE AND TEMPERATURE CONDITIONS

Source	Design Temperature	Design Pressure
Hot wells	121°F	3.5 in. Hg abs
Condensate pump discharge	150°F	600 psig
Polisher outlet	120°F	600 psig
Heater drain pump discharge and HD Tank	400°F	600 psig
Equalization Header[a]	400°F	200 psig
Steam generator feedwater	500°F	1800 psig
Main steam	600°F	1200 psig
Auxiliary condenser outlet	121°F	3.5 in Hg abs
Main steam after MSR	390°F	220 psia
Condensate storage tank	150°F	14.7 psia
Atmospheric drain tank #1	120°F	14.7 psia
Demineralized water	150°F	150 psig
Steam generator blowdown	130°F	300 psig

a) The Heater Drain Tank Equalization Header sample can be obtained by selective alignment of valves in the two Heater Drain sampling lines. Although either sample may be selected at any time, Chemistry typically will check the Equalization Header sample only during HD pump startup and prior to forward flowing Heater Drains to the Condensate System.

TABLE 10.4-19
EQUIPMENT DESIGN CRITERIA

(Sheet 1 of 2)

1. Hot Well Sample Pumps

Number of units	Two
Type horizontal centrifugal	
Operating mode, per unit	2 x 100%
Design flow, gpm	5
Design TDH, psi	32
NPSH available, psi	2
Casing design pressure, psig at 150°F	150
Material	type 316 stainless steel

2. Chilled Water Return Pump

Number of units	two
Type horizontal centrifugal	
Safety Class	NNS
Operating mode, per unit	1 x 100%
Design flow, gpm	15
Design TDH, psi	45
Normal TDH, psi	50
NPSH available, psi	5
Casing design pressure and temperature,	
psig at 150°F	125
Material	carbon steel

TABLE 10.4-19
EQUIPMENT DESIGN CRITERIA

(Sheet 2 of 2)

3.	Sample Coolers	
	Safety class	NNS
	Operating mode, percent each	100
	Tube side design flow, ml/min	500 to 2500
	Tube side design pressure and temperature	See Table 10.4-18
	Shell side design flow, gpm each	5
	Shell side design pressure, psig at 110°F	150
	Tube side temperatures	
	Inlet	varies per Table 10.4-18
	Outlet, °F	120
	Shell side temperatures	
	Inlet, °F	110
	Outlet, °F	160
	Heat exchanger type	shell-in-tube
	Material	
	Tube	stainless steel
	Shell	carbon steel

Note:

Total dynamic head (TDH); net positive suction head (NPSH); non-nuclear-safety-related (NNS)

TABLE 10.4-20
SECONDARY PLANT SAMPLING SYSTEM MEASURED PARAMETERS

(Sheet 1 of 3)

Sample Source	Parameter Measured	Receiver and Function
Hotwells (2 samples from 6 locations)	cation conductivity	recorder with high alarm
	sodium ion	recorder with high alarm
	anion/cation	computer database
Condensate pump discharge	pH	Note 1
	sodium ion	recorder with high alarm
	hydrazine	recorder with high and low alarms
	cation conductivity	recorder with high alarm
	specific conductivity	reorder
	anion/cation	computer database
	Secondary Corrosion Products	local sample panel monitored by Chemistry personnel
Heater drain pump discharge and HD Tank	cation conductivity	recorder with high alarm
Equalization Header[a]	anion/cation	computer database
	Secondary Corrosion Products	local sample panel monitored by Chemistry personnel
Polisher outlet	cation conductivity	recorder with high alarm
	sodium ion	recorder with high alarm
	anion/cation	computer database

TABLE 10.4-20
SECONDARY PLANT SAMPLING SYSTEM MEASURED PARAMETERS

(Sheet 2 of 3)

Sample Source	Parameter Measured	Receiver and Function
Final feedwater	hydrazine	recorder with high and low alarms
		recorder
	specific conductivity	recorder with high alarm
	sodium ion	recorder with high alarm
	cation conductivity	Note 1
	pH	
	anion/cation	computer database
	Secondary Corrosion Products	local sample panel monitored by Chemistry personnel
Vacuum deaerator outlet	dissolved oxygen	recorder with high alarm
Condensate storage tank	dissolved oxygen	recorder with high alarm
	cation conductivity	recorder with high alarm
	sodium ion	Local sample panel monitored by Chemistry personnel
Main Steam after MSR	cation conductivity	recorder with high alarm
	dissolved oxygen	recorder
	sodium ion	recorder with high alarm
	anion/cation	computer database
Atmosphere Drain Tank #1	cation conductivity	recorder with high alarm

TABLE 10.4-20
SECONDARY PLANT SAMPLING SYSTEM MEASURED PARAMETERS

(Sheet 3 of 3)

Sample Source	Parameter Measured	Receiver and Function
Demineralized Water Storage Tank	specific conductivity	recorder with high alarm
Steam Generator Blowdown Heat Exchanger outlet	Secondary Corrosion Products Secondary Chemistry Parameters as needed	local sample panel monitored by Chemistry personnel

a) The Heater Drain Tank Equalization Header sample can be obtained by selective alignment of valves in the two Heater Drain sampling lines. Although either sample may be selected at any time, Chemistry typically will check the Equalization Header sample only during HD pump startup and prior to forward flowing heater Drains to the Condensate System.

Note 1: The pH sample uses a common sensor selectable to one of the 2 sample sources. Chemistry typically will select the condensate sample during plant startup and select the feedwater sample during normal operation with indication via a recorder with high and low alarms.

11.0 RADIOACTIVE WASTE MANAGEMENT

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11.1 SOURCE TERMS

Source terms and models used in the design and evaluation of the waste processing systems are compared to operating plant data where available [1].

Source term models are presented which have been used for plant shielding design and effluent release analysis. Source terms for shielding design were based on design basis reactor coolant activity [2] assuming 1 percent fuel defects. Source terms for the effluent release analysis were based on the realistic model for reactor coolant activity as formulated in a standard for the American Nuclear Society (ANS) [3] and calculated per NUREG-0017[6].

Tritium production and fuel operating experience are fully addressed in References [1] and [4], respectively. For conservatism in the shielding design, radioactive waste streams are assumed to be recycled in the plant.

11.1.1 DESIGN BASIS MODEL FOR REACTOR COOLANT ACTIVITY

Except as noted, the discussion on radiation source terms presented in this section represents information used by the original license application to establish plant shielding and radwaste effluent assessments. It is retained for historical purposes to avoid loss of original design basis and is not subject to future update.

The parameters used in calculating the reactor coolant inventory at the SPU power level is presented in [New] Table 11.1-1A and the concentrations presented in [New] Table 12.2-3A. (Comment: These tables should be obtained from Westinghouse.)

Fission Products

The parameters used in the calculation of the reactor coolant fission product concentrations are summarized in Table 11.1-1 Concentrations are presented in Table 12.2-3.

The fission product concentrations are computed using the following differential equations:

For parent nuclides in the coolant:

$$\frac{dN_{wi}}{dt} = D\upsilon_i NC_i - \left(\lambda i + R\eta_i + \frac{B'}{B_o - tB'}\right)N_{wi}$$

For daughter nuclides in the coolant:

$$\frac{dN_{wj}}{dt} = D\upsilon_j NC_j - \left(\lambda j + R\eta_j + \frac{B'}{B_o - tB'}\right)N_{wj} + \lambda_i N_{wi}$$

where

N = population of nuclides (atoms)

t = time (sec)

D = clad defects, as a fraction of rated core thermal power being generated by rods with clad defects

R = purification flow (coolant system volumes per second)

B_o = initial boron concentration (ppm)

B' = boron concentration reduction rate by feed and bleed (ppm per second)

η = removal efficiency of purification cycle for nuclide

λ = radioactive decay constant (sec^{-1})

ν = escape rate coefficient for diffusion into coolant (sec^{-1})

subscripts

C = refers to core

w = refers to coolant

i = refers to parent nuclide

j = refers to daughter nuclide

The fission products are removed by decay, cleanup in the Chemical and Volume Control System and by letdown to the Boron Recycle System. No degassing is assumed in the volume control tank.

Corrosion Products

The corrosion product activities, which are independent of fuel defect level, are based on measurements at operating reactors [1]. The corrosion product concentrations are given in Table 12.2-3.

11.1.2 REALISTIC MODEL FOR REACTOR COOLANT ACTIVITY

The nominal plant design and operating parameters utilized by the referenced standard [3], together with the corresponding specific plant related parameters, are given in Table 11.1-3. The range of plant parameters used by the PWR-GALE computer code [6] is also provided. Corrections are made where a parameter falls outside the given range as noted in Reference [6]. The Gaseous Waste Processing System is assumed to strip fission gases from the volume control tank. The overall Y parameter, as given in Reference [3], is interpreted as being equivalent to the stripping fractions. A separate value of the stripping fraction for isotopes is used as indicated in Table 11.1-3. A stripping efficiency of 0.4 is used. This assumes a less

efficient separation in the volume control tank which results in a conservative estimate of the Reactor Coolant System radioactivity.

The stripping fractions and stripping efficiencies are related as follows:

$$SE = \frac{C_R - C_L}{C_R - C_{L_{eq}}}$$

$$SF = \frac{C_R - C_L}{C_R}$$

where

SE	=	stripping efficiency
SF	=	stripping fraction
C_R	=	gas concentration in the liquid phase entering the volume control tank
C_L	=	gas concentration in the liquid phase leaving the volume control tank
$C_{L_{eq}}$	=	gas concentration in the liquid phase leaving the volume control tank, assuming the ratio of the gas concentration in the vapor and liquid phases in the volume control tank follows Henry's Law.

Specific activities in the primary coolant, based on the realistic model with corrections for plant specific parameters, are given in Table 11.1-4.

Leakage from the primary system to the secondary side in the steam generator is possible; therefore, secondary side activity has been evaluated. The activity levels for the secondary side are presented in Table 11.1-7.

11.1.3 SOURCE TERMS FOR SHIELDING DESIGN

Liquid Waste Processing System

The source terms which were used for plant shielding design are based upon the coolant activity described in Section 11.1.1. However, for shielding purposes, the reactor coolant was assumed to be degassed. Shielding design source terms are presented for components outside the Containment which may contain reactor coolant or higher concentration of radioactivity. Source terms which have been used for shielding design are given in Table 11.1-5.

Gaseous Waste Processing System

The major components are the gas compressors, the hydrogen recombiners and the gas decay tanks. The gases are continuously recirculated; hence, the radiation sources in each component are identical. The total gaseous content is based on 40 years storage and stripping of the reactor coolant during reactor shutdown. The reactor coolant activity in Table 12.2-3 and 100 percent

stripping efficiency were assumed for shielding design. The resulting source terms are given in Table 11.1-5.

11.1.4 SOURCE TERMS FOR COMPONENT FAILURE

Updated source terms obtained from Reference [5] are used for plant shielding design review and to assess consequences of postulated component failure. The updated source terms represent results of improved calculational techniques and nuclear data bases.

Liquid Waste Processing System

The isotopic inventory of the 30,000 gallon floor drain tank no. 3 is given in Table 15.7-3. The tank inventory is based on updated reactor coolant activities obtained from Reference [5]. The capacity of the tank, and type of waste collected, show that analyzing the failure of this component is the most conservative assumption regarding the Liquid Waste Processing System.

Gaseous Waste Processing System

The isotopic inventories in one gas decay tank to be used for gas decay tank rupture are given in Table 15.7-2. The inventories are based on updated reactor coolant activities obtained from Reference [5] and assuming 0.7 scfm volume control tank purge rate to define the maximum activities. The activity is further based on 40 years inventory of the Kr-85 and equilibrium levels of all other isotopes. It is assumed two units are operating simultaneously and decay tanks are switched regularly.

REFERENCES

1. “Source Term Data for Westinghouse Pressurized Water Reactors,” WCAP-8253, Amendment 1, July 1975.

2. “Radiation Analysis Design Manual, 4 Loop Plant,” WCAP-7664, Revision 1, October 1972.

3. American National Standard Radioactive Source Term for Normal Operation of Light Water Reactors, ANSI/AnS-18.1-1984.

4. “Operational Experience with Westinghouse Cores,” WCAP-8183, Revision 6, June 1977.

5. “Radiation Analysis Manual, Standard Plant Model 412,” Revision 3 Westinghouse Electric Corporation, November 1978.

6. “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code),” NUREG-0017, Revision 1, 1985.

TABLE 11.1-1
PARAMETERS USED IN THE CALCULATION OF DESIGN BASIS PRIMARY COOLANT ACTIVITIES - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 3)

1.	Ultimate core thermal power (MWt)	3565
2.	Clad defects, as a percent of rated core thermal power being generated by rods with clad defects	1.0
3.	Reactor coolant liquid volume (ft^3)	12,000
4.	Reactor coolant full power average temperature (°F)	590
5.	Purification flow rate, normal (gpm[b])	75
6.	Effective cation demineralizer flow (gpm[b])	7.5
7.	Volume control tank volumes	
	a. Vapor (ft^3)	240
	b. Liquid (ft^3)	160
8.	Fission product escape rate coefficients[c]	
	a. Noble gas isotopes (sec^{-1})	6.5×10^{-8}
	b. Br, Rb, 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, Zr, Nb, La, Ce, Pr isotopes (sec^{-1})	1.6×10^{-12}
9.	Mixed bed demineralizer decontamination factors	
	a. Noble gases and Cs-134, 136, 137, Y-90, 91 and Mo-99	1.0
	b. All other isotopes including corrosion products	10.0

TABLE 11.1-1
PARAMETERS USED IN THE CALCULATION OF DESIGN BASIS PRIMARY COOLANT ACTIVITIES - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 3)

10. Cation bed demineralizer decontamination factor for Cs-134, 137, Y-90, 91 — 10.0

11. Volume control tank noble gas stripping fraction

Isotope	Stripping Fraction: Assuming Operation of GWPS[d]	Stripping Fraction: Plant Without GWPS
Kr-85	2.3×10^{-1}	2.3×10^{-5}
Kr-85m	2.9×10^{-1}	2.7×10^{-1}
Kr-87	6.0×10^{-1}	6.0×10^{-1}
Kr-88	4.3×10^{-1}	4.3×10^{-1}
Xe-131m	2.5×10^{-1}	1.0×10^{-2}
Xe-133	2.5×10^{-1}	1.6×10^{-2}
Xe-133m	2.6×10^{-1}	3.7×10^{-2}
Xe-135	2.8×10^{-1}	1.8×10^{-1}
Xe-135	8.0×10^{-1}	8.0×10^{-1}
Xe-138	1.0	1.0

12. Partition factor[e] for iodine in the volume control tank — 100.0

13. Boron concentration and reduction rates

a. B_o, initial cycle, (ppm) — 805
   B^1, initial cycle, (ppm/day) — 2.06

b. B_o, equilibrium cycle, (ppm) — 1080
   B^1, equilibrium cycle, (ppm/day) — 3.96

TABLE 11.1-1
PARAMETERS USED IN THE CALCULATION OF DESIGN BASIS PRIMARY COOLANT ACTIVITIES - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 3)

14.	Pressurizer volumes		
	a.	Vapor (ft^3)	720
	b.	Liquid (ft^3)	1080
15.	Spray line flow (gpm[b])		1.0
16.	Pressurizer stripping fractions		
	a.	Noble gases	1.0
	b.	All other elements	0

a) Historical, not subject to update; has been retained to preserve original design basis. Parameters used to develop the SPV RCS source terms presented in Table 11.1-1A.

b) At reference condition, 130°F and 2300 psig, 1 gpm = 493.6 lb/hr.

c) Escape rate coefficients are based on fuel defect tests performed at the Saxton Reactor. Recent experience at plants operating with fuel rod defects has verified the listed escape rate coefficients

d) Volume control tank purge rate is 0.7 scfm and volume control tank stripping efficiency is 40 percent.

e) Refer to Reference [1].

TABLE 11.1-2
TABLE 11.1-2 HAS BEEN DELETED.

TABLE 11.1-3
PARAMETERS USED IN THE CALCULATION OF REALISTIC PRIMARY COOLANT ACTIVITIES ORIGINAL LICENSING BASIS[(a)]

(Sheet 1 of 3)

Parameter	Symbol	Units	Nominal Value	Range		
				Maximum	Minimum	CPNPP
Thermal Power	P	MWT	3400	3800	3000	3565
Steam flow rate	FS	lb/hr	$1.5x10^7$	$1.7x10^7$	$1.3x10^7$	$1.5x10^7$
Weight of water in Reactor Coolant System	WP	Lb	$5.5x10^5$	$6.0x10^5$	$5.0x10^5$	$5.1x10^5$
Weight of water in all steam generators	WS	Lb	$4.5x10^5$	$5.0x10^5$	$4.0x10^5$	$3.84 x10^5$
Reactor coolant letdown flow (purification)	FD	lb/hr	$3.7x10^4$	$4.2x10^4$	$3.2x10^4$	$3.8x10^4$
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/hr	500	1000	250	100
Steam generator blowdown flow (total)	FBD	lb/hr	$7.5x10^4$	$1.0x10^5$	$5.0x10^4$	$6.96 x10^4$
Fraction of radioactivity in blowdown stream which is not returned to secondary coolant system	NBD	-	1.0	1.0	0.9	1.0
Flow through the purification system cation demineralizer	FA	lb/hr	3700	7500	0.0	$3.8x10^3$
Ratio of condesate demineralizer flow rate to the total steam flow rate	NC	-	0.0	0.01	0.0	0.7

TABLE 11.1-3
PARAMETERS USED IN THE CALCULATION OF REALISTIC PRIMARY COOLANT ACTIVITIES ORIGINAL LICENSING BASIS[(a)]

(Sheet 2 of 3)

Ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount of noble gases routed to the primary coolant system from the purification system (not including the Boron Recycle System)	Y	-	0.0	0.01	0.0	See below

Isotope	Y Parameter
Kr-83m	0.37
Kr-85m	0.34
Kr-85	0.28
Kr-87	0.38
Kr-88	0.36
Kr-89	0.40
Xe-131m	0.25
Xe-133m	0.26
Xe-133	0.25
Xe-135m	0.39
Xe-135	0.30
Xe-137	0.40
Xe-138	0.39

TABLE 11.1-3
PARAMETERS USED IN THE CALCULATION OF REALISTIC PRIMARY COOLANT ACTIVITIES ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 3)

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

TABLE 11.1-4
REALISTIC REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 3)

Isotope	Reactor Coolant Activity (μCi/gram)	Isotope	Reactor Coolant Activity (μCi/gram)
Group I - Noble Gases		Group III - Cs, Rb	
Kr-83m	2.13×10^{-2}		
Kr-85m	1.03×10^{-1}		
Kr-85	7.12×10^{-3}		
Kr-87	6.20×10^{-2}	Cs-134	7.10×10^{-3}
Kr-88	1.97×10^{-1}	Cs-136	8.70×10^{-4}
Kr-89	5.41×10^{-3}	Cs-137	9.40×10^{-3}
Xe-131m	1.89×10^{-2}		
		Group IV - N-16[b]	
Xe-133m	1.03×10^{-1}		
Xe-133	5.15	N-16	4.00×10^{1}
Xe-135m	1.40×10^{-2}		
Xe-135	2.98×10^{-1}	Group V - Tritium[b]	
Xe-137	9.73×10^{-3}		
Xe-138	4.73×10^{-2}	H-3	1.00
Group II - Halogens			
I-131	4.50×10^{-2}		
I-132	2.10×10^{-1}		
I-133	1.40×10^{-1}		
I-134	3.40×10^{-1}		

TABLE 11.1-4
REALISTIC REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 3)

Isotope	Reactor Coolant Activity (μCi/gram)	Isotope	Reactor Coolant Activity (μCi/gram)
I-135	2.60×10^{-1}		
Group VI - Other Isotopes		Group VI - Other Isotopes	
Na-24	4.70×10^{-2}		
Cr-51	3.10×10^{-3}	Te-131m	1.50×10^{-3}
Mn-54	1.60×10^{-3}	Te-131	7.70×10^{-3}
Fe-55	1.20×10^{-3}	Te-132	1.70×10^{-3}
Fe-59	3.00×10^{-4}		
Co-58	4.60×10^{-3}	Ba-140	1.30×10^{-2}
Co-60	5.30×10^{-4}	La-140	2.50×10^{-2}
Zn-65	5.10×10^{-4}		
Sr-89	1.40×10^{-4}	Ce-141	1.50×10^{-4}
Sr-90	1.20×10^{-5}	Ce-143	2.80×10^{-3}
Sr-91	9.60×10^{-4}	Ce-144	3.90×10^{-3}
		W-187	2.50×10^{-3}
Y-91m	4.60×10^{-4}		
Y-91	5.20×10^{-6}	Np-239	2.20×10^{-3}
Y-93	4.20×10^{-3}		
Zr-95	3.90×10^{-4}		
Nb-95	2.80×10^{-4}		
Mo-99	6.40×10^{-3}		
Tc-99m	4.70×10^{-3}		

TABLE 11.1-4
REALISTIC REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 3)

Isotope	Reactor Coolant Activity (μCi/gram)	Isotope	Reactor Coolant Activity (μCi/gram)
Ru-103	7.50×10^{-3}		
Ru-106	9.00×10^{-2}		
Ag-110m	1.30×10^{-3}		
Te-129m	1.90×10^{-4}		
Te-129	2.40×10^{-2}		

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

b) Values provided from ANSI/ANS-18.1-1984 [3]

TABLE 11.1-5
SOURCE TERMS FOR SHIELDING DESIGN SHIELDING SOURCES WASTE PROCESSING SYSTEM - ORIGINAL LICENSING BASIS[(a)]

(Sheet 1 of 6)

Waste Evaporator Condensate Demineralizer

($30\ ft^3$ at Resin)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.4	2.6×10^4
0.8	2.0×10^4
1.3	3.5×10^3
1.7	1.7×10^3
2.2	6.2×10^2

Waste Monitor Tank Demineralizer

($30\ ft^3$ at Resin)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.4	1.1×10^6
0.8	3.0×10^6
1.3	9.9×10^5
1.7	4.8×10^5
2.2	1.8×10^5

TABLE 11.1-5
SOURCE TERMS FOR SHIELDING DESIGN SHIELDING SOURCES WASTE PROCESSING SYSTEM - ORIGINAL LICENSING BASIS[(a)]

(Sheet 2 of 6)

Waste Holdup Tank, Floor Drain Tank, and Waste Monitor Tank

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.4	3.6×10^4
0.8	2.0×10^5
1.3	1.3×10^5
1.7	5.0×10^4
2.2	1.9×10^4

Spent Resin Storage Tank

($350\ ft^3$ at Resin)

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.4	1.5×10^8
0.8	3.4×10^8
1.3	3.3×10^7
1.7	1.7×10^7
2.2	4.5×10^6
2.5	1.0×10^5
3.5	2.0×10^5

TABLE 11.1-5
SOURCE TERMS FOR SHIELDING DESIGN SHIELDING SOURCES WASTE PROCESSING SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 6)

Evaporator Concentrates[b]

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.8	1.2×10^{6}

Drumming Station Spent Resin

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.8	1.0×10^{8}
1.3	1.0×10^{7}

Waste Monitor Tank Filter[c]

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.8	3.4×10^{7}
1.3	8.9×10^{6}

TABLE 11.1-5
SOURCE TERMS FOR SHIELDING DESIGN SHIELDING SOURCES WASTE PROCESSING SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 4 of 6)

Waste Evaporator Condensate Filter[c]

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.4	1.2×10^5
0.8	2.2×10^5
1.3	2.5×10^4

Waste Evaporator Vent Condenser Vapor[c]

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.1	1.1×10^7
0.4	3.5×10^6
0.8	1.5×10^6
1.7	1.0×10^6
2.2	3.8×10^6
2.5	5.2×10^6

TABLE 11.1-5
SOURCE TERMS FOR SHIELDING DESIGN SHIELDING SOURCES WASTE PROCESSING SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 5 of 6)

Hydrogen Recombiner, Waste Gas Compressor,and Gas Decay Tanks

Gamma Energy (Mev/)	Specific Source Strength (Mev/cc-sec)
0.1	1.8×10^{6}
0.4	3.4×10^{5}
0.8	8.6×10^{4}
1.7	6.7×10^{4}
2.2	1.2×10^{5}
2.5	2.9×10^{5}

Waste Evaporator Feed Filter, and Floor Drain Tank Filter[d]

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.8	3.4×10^{7}
1.3	8.9×10^{6}

TABLE 11.1-5
SOURCE TERMS FOR SHIELDING DESIGN SHIELDING SOURCES WASTE PROCESSING SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 6 of 6)

Spent Resin Sluice Filter[d]

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.8	1.1×10^7
1.3	3.0×10^6

a. Historical, not subject to future updating. Has been retained to preserve original design basis.

b. 1.89×10^6 grams - maximum capacity for evaporator concentrate.

c. Homogeneous sources with the following dimensions and compositions

	Source Dimensions(in)		Source Composition
Component	Radius	Length	(Volume Percent)
Waste Monitor Tank, Waste Evaporator Condensate Filters	1.25	19	30% air, 70% water
Waste Evaporator Vent Condenser	4	20	30% s/s, 22% water, 48% air

d. Homogeneous sources with the following dimensions and compositions:

	Source Dimensions(in)		Source Composition
Component	Radius	Length	(Volume Percent)
Waste Evaporator Drain Tank	1.25	19	30% air, 70% water
Spent Resin Sluice	3.375	19	62% air, 38% water

TABLE 11.1-6
TABLE 11.1-6 HAS BEEN DELETED.

TABLE 11.1-7
REALISTIC SECONDARY SIDE (WATER) EQUILIBRIUM ACTIVITIES[a] -
ORIGINAL LICENSING BASIS[b]

(Sheet 1 of 3)

Isotope	Concentration (uCi/gm)
H-3	1.00E-03[c]
N-16	1.00E-06[c]
Na-24	1.55E-06
Cr-51	1.30E-07
Mn-54	6.51E-08
Fe-55	4.91E-08
Fe-59	1.20E-08
Co-58	1.90E-07
Co-60	2.20E-08
Zn-65	2.10E-08
Sr-89	5.71E-09
Sr-90	4.91E-10
Sr-91	2.93E-08
Y-91	2.11E-10
Y-91m	3.65E-09
Y-93	1.25E-07
Zr-95	1.60E-08
Nb-95	1.10E-08
Mo-99	2.53E-07
Tc-99m	1.17E-07

TABLE 11.1-7
REALISTIC SECONDARY SIDE (WATER) EQUILIBRIUM ACTIVITIES[a] -
ORIGINAL LICENSING BASIS[b]

(Sheet 2 of 3)

Isotope	Concentration (uCi/gm)
Ru-103	3.11E-07
Ru-106	3.71E-06
Ag-110M	5.31E-08
Te-129	2.49E-07
Te-129m	7.82E-09
Te-131	3.35E-08
Te-131m	5.51E-08
Te-132	6.66E-08
I-131	1.44E-06
I-132	3.10E-06
I-133	4.02E-06
I-134	2.59E-06
I-135	5.96E-06
Cs-134	3.05E-07
Cs-136	3.71E-08
Cs-137	4.07E-07
Ba-140	5.22E-07
La-140	9.45E-07
Ce-141	6.12E-09
Ce-143	1.02E-07
Ce-144	1.60E-07

TABLE 11.1-7
REALISTIC SECONDARY SIDE (WATER) EQUILIBRIUM ACTIVITIES[a] -
ORIGINAL LICENSING BASIS[b]

(Sheet 3 of 3)

Isotope	Concentration (uCi/gm)
W-187	8.91E-08
Np-239	8.50E-08

a) Equilibrium activity is based on the following:

1. Primary coolant leak - 75 lb/day
2. Primary coolant activity - in accordance with Table 11.1-4
3. Steam generator blowdown rate - 6.96 x 10^4 lb/hr
4. Ratio of condensate demineralizer flow to steam flow rate - 0.70
5. Decontamination Factors

	Blowdown	Condensate Demineralizer
Cs, RB	100	1
Anions	100	1
Others	1000	1

b) Historical, not subject to future updating. Has been retained to preserve original design basis.

c) Values provided from ANSI/ANS - 18.1-1984

11.2 LIQUID WASTE MANAGEMENT SYSTEM

The CPNPP Liquid Waste Processing System (LWPS) services both units with shared components as described in Section 1.2.2.12 of the FSAR. The system is classified as non-nuclear safety-related.

The radioactivity values presented in this section are the design basis values used for the design of the liquid waste management system. As such they are considered historical and are not subject to future updating. The information is retained to avoid loss of original design basis. Actual radioactivity release quantities can be found in the annual radioactive effluent release reports submitted to the NRC.

11.2.1 DESIGN BASES

11.2.1.1 Design Objective

The LWPS is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences as follows:

1. LWPS equipment malfunction

2. Excessive leakage in Reactor Coolant System (RCS) equipment

3. Excessive leakage in auxiliary system equipment

These modes of operation are considered in combination with fuel cladding defects of 1.0 percent.

Table 11.2-8 shows the design inventory of radionuclides in LWPS components.

The system design considers potential population and occupational exposures and ensures that quantities of radioactive releases to the environment meet the requirements specified in 10 CFR Parts 20 and 50 and the dose design objectives specified in Appendix I of 10 CFR Part 50, during both normal and anticipated operational occurrences.

11.2.1.2 Design Criteria

1. Release limits

   The following summarizes the basic radioactive liquid release limits established by 10 CFR Part 20[a]:

   a. The concentration limit on an unidentified, instantaneous release basis is as defined in 10 CFR Part 20, Appendix B.

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a. Design criteria based on the provisions of 10 CFR 20.1-20.601.

b. The concentration limit on an identified basis is defined in 10 CFR Part 20, Appendix B, Table II, Column 2.

These concentration limits are considered at the point of discharge to Squaw Creek Reservoir. For purposes of calculating doses from liquid effluents, the concentration within the exclusion area boundary may be determined by applying appropriate factors for dilutions, dispersion, and decay between the point of discharge and the exclusion area boundary.

2. Dose to Individuals

Except for occasional variation, as specified in 10 CFR Part 20[a], the permissible dose of exposure for an individual is as follows:

a. Restricted Area

1 1/4 rems per calendar quarter to the whole body

b. Unrestricted Area

0.5 rem per calendar year to the whole body

A cost-benefit analysis for the LWPS was not required because the limitations imposed by the annex Appendix I of 10 CFR Part 50 were met by the LWPS as described in the FSAR during construction.

Evaluations showing that the proposed systems are capable of controlling releases within the numerical design objectives of Appendix I of 10 CFR Part 50 may be found in Appendix 11A.

11.2.1.3 Release Control Capability

The system is capable of controlling release and reducing doses. The subsystems have components such as filters, evaporators and demineralizers, all of which remove radioactivity to various degrees.

The system is capable of recycling water back to plant systems.

The decontamination factors for the processing equipment, which were used to determine isotopic release quantities, are listed in Table 11.2-6.

The surge capacity of the LWPS permits the system to accommodate waste until failures can be fixed and normal plant operation is resumed.

11.2.1.4 Seismic Design Classification and Quality Group Classification

The seismic design classification and quality group classification for the LWPS components and structures are listed in Appendix 17A, Table 17A-1.

11.2.1.5 Special Features

The system has been designed and the equipment chosen to effect an appreciable reduction in maintenance downtime, leakage, gas release, cleaning, etc.

The tanks have sufficient capacity to store abnormal surge loads. They are equipped with high-level alarms and low-level alarms to provide operator notification of a potential overflow condition or leaking tank.

A majority of the pumps are of the canned-rotor design. The canned-rotor design minimizes overall liquid leakage and also minimizes the release of entrained radioactive gases in the leaking fluid to the plant building atmosphere.

Provisions are made to stop radioactive releases by providing a stop valve interlocked with a radiation monitor on the system discharge line. The valve automatically closes when the waste activity reaches the monitor set point. As an additional precaution, the stop valve is also interlocked to block flow if both units have less than two circulating water pumps running.

11.2.2 SYSTEM DESCRIPTIONS

11.2.2.1 General

The LWPS collects and processes potentially radioactive wastes for recycle or disposal during the normal mode of operation. Provisions are made to sample and analyze fluids before they are discharged. Based on this analysis, these wastes are either released under controlled conditions via the circulating water discharge canal or retained for further processing. The circulating waterflow serves to reduce the concentration of radioactivity in the plant effluent by diluting the LWPS discharges. A permanent record of radioactivity releases is provided by analyses of known volumes of effluent.

Normally, radioactive liquids discharged from the RCS are recycled or processed by the Boron Recycle System (BRS), thereby limiting inputs into the LWPS. Water in the Recycle Holdup Tank that needs to be processed is sent to the Filter/Demineralizer System (FDS).

The original system design intent was for Comanche Peak to be a “no release” plant, thereby recycling and reusing all effluent streams. For ALARA reasons, this philosophy has changed and the intent now is to prevent any tritiated or otherwise contaminated effluent from reaching the Reactor Makeup Water Storage Tanks. For this reason the isolation valves between the WP Waste Evaporator Condensate Pumps and the DD/RMW System are presently administratively controlled in a closed position. Also being maintained closed in the BR System is the upstream isolation valve to the radiation controlled three-way diversion valve. In any case, previously designed processing of effluents is possible by simply opening these valves and aligning other valves as appropriate.

11.2.2.2 Subsystems

The LWPS is designed to segregate different effluents from equipment leaks and drains according to their chemical and radiochemical properties. In addition interconnecting piping is available to allow for operating flexibility and provide for efficient utilization of purification equipment. The system is divided into the following subsystems:

1. Reactor Coolant Drain Tank

2. Drain Channel A

3. Drain Channel B

4. Drain Channel C

5. In addition, the LWPS provides capability for handling and storage of spent ion-exchange resins

6. Drain systems are also provided inside the Containment, Fuel, Auxiliary, and Safeguards Buildings to collect drains and leaks and to transfer them to appropriate tanks

7. A filter/demineralizer system (FDS) is typically used to process liquid waste. This filter/demineralizer system is located in Fuel Building Room X-247 on elevation 800'-2".

8. An interim/preprocessing drum dryer system may be used for conditioning or volume reduction of non-routine liquid radioactive water generated from activities like tank cleaning, facility/equipment decontamination, or tank bottoms. This system is a self-contained skid mounted system located in the Fuel Building.

The LWPS subsystems have process capacities as outlined in the following paragraphs.

In the event of equipment faults of moderate frequency, the LWPS is capable of processing up to one-gpm of primary coolant leakage with no change in system operation. During these equipment faults, the load on the waste processing system is increased.

As a practical upper limit of system operation, the LWPS can process 50 gpm.

It is possible to operate the LWPS at these upper limit processing rates for extended periods. However, close operator surveillance is required, and radioactive discharge must be kept within the as low as reasonably achievable (ALARA) limit. Assuming the discharge limits are not exceeded and there are no equipment malfunctions, the LWPS can process all anticipated inputs to the system.

Instrumentation and controls necessary for the operation of the LWPS are located on control boards in the Auxiliary Building and Fuel Building. Any alarm on these control boards is relayed to the main control boards in the Control Room.

1. Reactor Coolant Drain Tank Subsystem

   Recyclable reactor-grade effluents enter this subsystem from equipment leaks and drains, valve leakoffs, pump seal leakoffs, loop drain leakoffs, and from other deaerated tritiated water sources inside the Containment. Connections are provided for various drains and leakoffs and for cooling the Pressurizer Relief Tank. This deaerated tritiated liquid is normally pumped directly to the Recycle Holdup Tanks via the Reactor Coolant Drain Tank Heat Exchanger. This liquid may be processed by the BRS rather than by the LWPS. In addition, refueling canal drains can be routed to the Spent Fuel Purification System using the Reactor Coolant Drain Tank pumps.

The Reactor Coolant Drain Tank is also connected to the Gaseous Waste Processing System vent header. Except in times immediately prior to a planned unit shutdown, hydrogen gas is normally supplied to the Reactor Coolant Drain Tank to maintain a hydrogen blanket and limit hydrogen usage. Provisions for sampling gas are provided. Details of the Reactor Coolant Drain Tank Subsystem are shown on Figure 11.2-2.

2. Drain Channel A Subsystem

Aerated tritiated liquid enters Drain Channel A through lines connected to the Waste Holdup Tank. Sources of this aerated liquid are as follows:

a. Accumulator drainage (to Reactor Coolant Drain Tank Pump suction)

b. Sample room sink drains (to Waste Holdup Tank);

c. Ion exchanger, filter, pump, and other equipment drains (to Waste Holdup Tank)

The Waste Holdup Tank is one of the initial collecting points for liquid waste to be processed through the waste evaporator or the FDS.

Abnormal liquid sources include leaks which may develop in the RCS and its auxiliary systems. Considerable surge and processing capacity is incorporated in the LWPS to accommodate abnormal operations.

The basic composition of the liquid collected in the Waste Holdup Tank is boric acid and water with some radioactivity.

The Waste Evaporator Feed Pump delivers the contents of the Waste Holdup Tank through a filter to the Waste Evaporator Package, the FDS or Floor Drain Tank 3 for removal of radioisotopes and boron prior to reuse or discharge.

If the waste evaporator is processing liquid waste, the condensate leaving the waste evaporator can be routed either directly to a tank, or filtered and then routed to a tank, or demineralized then filtered and then routed to a tank. The final disposition of the condensate is typically dependent on the radioactivity levels found in samples taken from the process stream. Based upon sample results, the condensate in the tanks can then be transferred to a monitor tank and discharged, or be processed through the FDS. The evaporator bottoms may be packaged or processed.

The effluent from the FDS will be directed to one of the six monitor tanks.

The waste evaporator or the FDS may also be utilized as a backup to the Boron Recycle Evaporator to process Recycle Holdup Tank contents.

Details of the Drain Channel A subsystem are shown on Figure 11.2-3.

3. Drain Channel B Subsystem

Drain Channel B collects and processes non-reactor-grade liquid wastes. These include floor drains, equipment drains containing non-reactor-grade water, and other non-reactor-grade sources.

Drain Channel B equipment includes three floor drain tanks, a common filter and evaporator, two waste monitor tanks with a common demineralizer and filter, and two additional monitor tanks.

Three floor drain tanks (numbered 1, 2, and 3) are provided to collect liquid waste from both units. Floor Drain Tanks 1 and 2 each have a capacity of 10,000-gal. Floor Drain Tank 1 normally collects waste water from the Unit 1 and common sources. Floor Drain Tank 2 normally collects waste water from the Unit 2 sources. Floor Drain Tank 3 has a 30,000-gal. capacity. Although any of the Floor Drain Tanks can be directly processed, Floor Drain Tank 3 is normally used as a batch tank for processing Floor Drain Tanks 1 and 2. In addition, Floor Drain Tank 3 is an alternate liquid waste collector for some of the common sources.

Laboratory samples which contain reagent chemicals (and possibly tritiated liquid) are sent to the Chemical Drain Tank. Rinse water from the laboratory is discarded to either the Floor Drain Tanks or the Chemical Drain Tank.

When Floor Drain Tank 3 is sufficiently full, the contents are normally sampled and analyzed to determine the type of processing required. The capability to recycle the waste is dependent on its chemical properties, water inventories and radioactivity level.

The processing flow paths for floor drain wastes are through the floor drain tank pump, strainer, filter, and evaporator; or through the FDS. Evaporator effluents may then be routed to a waste monitor tank. The evaporator bottoms are disposed of as required. Floor drain evaporator or FDS process effluents are collected in either the Waste Monitor Tanks, the Laundry Holdup & Monitor Tanks or the Plant Effluent Holdup and Monitor Tanks. Processed fluid in a monitor tank is mixed and sampled. Depending on analyses, a processed batch of waste may be subject to controlled release or returned to Drain Channel B for additional treatment.

Any releases to the environment have radioactivity levels that are ALARA (within the limits of Appendix I of 10 CFR Part 50) and are administratively controlled by using a downstream stop valve interlocked with a radiation monitor. The monitor initiates automatic closure of the valve when the setpoint is exceeded. The radiation monitor setpoint is established at a value below the required limits to provide a margin of assurance that release limits will not be exceeded. A permanent record of radioactive releases is provided by an analysis of the known volumes of waste effluent released. The stop valve is also interlocked with a minimum of two circulating water pumps running.

Details of the Drain Channel B subsystem are shown on Figure 11.2-4.

4. Drain Channel C Subsystem

Drain Channel C is provided to collect and process waste effluents from onsite vendor laundry, personnel decontamination showers and sinks, and surface decontamination. These liquids may be collected in the Laundry and Hot Shower Tank. Drain Channel C

equipment includes a Laundry and Hot Shower Tank, strainer and filter, two Laundry Holdup and Monitor Tanks, and a Laundry Water Head Tank.

Based on operating plant data, the volume of laundry and hot shower waste is estimated to be approximately 20,000-gal per year per unit. Processing of contaminated plant laundry will be performed by an outside vendor.

The liquid collected in the Laundry and Hot Shower Tank is pumped through the Laundry and Hot Shower Tank Strainer and filter to one of the two 5000-gal Waste Monitor Tanks. The waste water is then sampled to determine if the liquid is to be discharged or reprocessed through the FDS or the waste evaporator. With the use of the FDS the Laundry Holdup Monitor Tanks may also receive effluent.

5. Spent Resin Handling Subsystem

This subsystem collects, handles, and processes spent resins from the primary fluid systems prior to their disposal.

Normally, resin from the primary system demineralizers is transported to and stored in the Spent Resin Storage Tank prior to being packaged for disposal. The Spent Resin Sluice portion of the LWPS consists of a Spent Resin Sluice Filter, Spent Resin Sluice Pump, and the Spent Resin Storage Tank. The resin sluice water, after being directed to an ion exchange vessel by the sluice pump, is returned to the spent resin storage tank for reuse. Thus, sluicing of spent resin from primary plant demineralizers is normally accomplished without generating a large volume of additional liquid waste.

For additional system flexibility, the Spent Resin Storage Tank may be bypassed and the resin sluiced directly from a demineralizer to a mobile system for disposal.

The resin slurry from the Spent Resin Storage Tank is transferred to the waste processing area by pressurizing the tank with nitrogen.

Resin from the Steam Generator Blowdown System (SGBS) demineralizers is handled in a similar manner as that described for the primary system demineralizers. The SGBS resin handling is physically separated from primary demineralizers and is discussed in Section 10.4.8. The primary system resins and the SGBS resins are segregated during all phases of handling so that no cross contamination is possible.

Powdered resin from the Condensate Cleanup System (CCS) is processed as described in Section 10.4.6. When specified by the Radiological Effluents Control Program the powdered resin will be disposed of as radioactive waste using vendor equipment. The actual process for the packaging of resins for disposal is discussed in Subsection 11.4.

Details of the spent resin handling subsystem are shown on Figures 11.2-6 and 11.2-7.

11.2.2.3 Liquids from Sources Other Than Liquid Waste Processing System

1. Steam Generator Blowdown

Blowdown from the steam generators of each unit is cooled, filtered, demineralized, and returned to the condenser or heater drain tank for reuse as secondary coolant. This Blowdown Processing System is described in Section 10.4.8.

2. Turbine Building Sumps

Discharges from the Turbine Building Sumps are routed to the Wastewater Management System (WMS), as discussed in Section 9.2.8. As shown on Figure 9.2-15, these discharges are normally routed to the Low Volume Waste treatment facilities. However, when radioactivity is present above specified levels, the discharges are diverted to the Co-Current Waste treatment facilities. These facilities are also part of the WMS. After batching, these wastes are sampled for radioactivity and if required, treated for conventional pollutants and discharged to the Circulating Water Discharge Canal.

11.2.2.4 Equipment Description and Design Basis

11.2.2.4.1 General

The locations of the LWPS components are shown in Table 11.2-4.

The Waste Holdup Tank, Floor Drain Tanks, and Laundry and Hot Shower Tank are located at the lowest possible levels of the plant so that gravity feed into the tanks is possible from the majority of the components being drained.

Typically parts or components in contact with borated water are fabricated from or clad with austenitic stainless steel. Except for flanged joints which are provided at certain equipment connections to reduce personnel exposure during maintenance or during replacement of that equipment, all-welded construction is used. Compression fittings will be used on small diameter instrument lines. The design parameters of this equipment are listed in Table 11.2-3.

Tank overflow protection is provided as described in item 9 of Section 11.2.2.4.2.

11.2.2.4.2 Equipment

1. Pumps

The two basic types of pumps in the LWPS are a canned-rotor pump design and a mechanical seal design.

Valves are installed in pump discharge lines. Design bases for individual pumps are given below:

a. Reactor Coolant Drain Tank Pump

In accordance with the design basis, this pump may perform as a plant drain. Two pumps are furnished because of the relative inaccessibility of the Containment during plant operation. One pump provides sufficient flow for normal operation of the Reactor Coolant Drain Tank portion of the LWPS. The Reactor Coolant Drain Tank Pumps are of the canned-rotor design.

b. Waste Evaporator Feed Pump

This standard design canned-centrifugal pump supplies feed to the Waste Evaporator or FDS based on the level in the Waste Holdup Tank.

Pump head requirements include elevation, the head loss through the feed filter, and the required waste evaporator feed supply pressure.

c. Waste Evaporator Condensate Pump

The Waste Evaporator Condensate Tank Pump is a canned-rotor design transfer pump. One pump is used to transfer the contents of the Waste Condensate Tank to the BRS holdup tank or to one of the Waste Monitor Tanks.

d. Floor Drain Tank Pumps

Mechanical-seal type pumps are supplied for Floor Drain Tank Pumps 1 and 2. Floor Drain Tank Pump 3 is a canned-rotor design pump. Each pump is capable of supplying feed at the required pressure to the Floor Drain Evaporator, the Waste Monitor Tank Demineralizers and the FDS.

e. Waste Monitor Tank Pumps

One canned-rotor pump is used for each of the two Waste Monitor Tanks. These pumps can discharge water from the plant or recycle it for further processing if required.

f. Laundry and Hot Shower Tank Pump

This pump is of the mechanical-seal type and has performance characteristics identical to the standard LWPS pump. The pump is designed to transfer water from the Laundry and Hot Shower Tank to the Waste Monitor Tanks via the strainer and filter.

g. Reverse Osmosis Concentrates Tank Pump

This pump has been abandoned in place. The Reverse Osmosis System will not be used.

h. Laundry Holdup and Monitor Tank Pump

These pumps are canned-rotor type. These pumps can discharge water from the plant or recycle it for further processing, if required.

i. Chemical Drain Tank Pump

One standard canned-rotor pump is used for recirculation and to transfer liquid from the Chemical Drain Tank to the Waste Conditioning Tank.

j. Spent Resin Sluice Pump

This pump is a canned-rotor design and operates during resin sluicing. The design flow rate is based on a conservative fluid velocity sufficient to sluice resin in a three-inch pipe.

k. Plant Effluent Holdup and Monitor Tank Pumps

Two horizontal centrifugal pumps provided with mechanical seals operate to permit recirculation, sampling, and controlled release or retreatment of processed waste in the floor drain waste system (Drain Channel B).

2. Tanks

a. Reactor Coolant Drain Tank

One tank is provided for each unit. The purpose of the Reactor Coolant Drain Tank is to collect leakoff-type drains inside the Containment at a central collection point for further disposition through a single penetration via the Reactor Coolant Drain Tank Pumps. The tank provides surge volume and net positive suction head (NPSH) to the pumps.

Only water which can be directed to the Boron Recycle Holdup Tanks enters the Reactor Coolant Drain Tank. The water must be compatible with reactor coolant and not contain dissolved air or nitrogen. A hydrogen gas blanket is normally maintained on the tank. Within approximately 24 hours of a scheduled unit shutdown, nitrogen may be introduced into the RCDT to purge the hydrogen in preparation for the shutdown.

It is intended that the tank be maintained at a constant level to minimize the amount of gas sent to the GWPS.

The level is maintained by running one pump continuously and installing a proportional control valve in the discharge line. This valve recirculates a portion of the pump discharge back to the Reactor Coolant Drain Tank to maintain the tank level.

Continuous flow is maintained through the heat exchanger in order to prevent loss of pump NPSH resulting from a sudden inflow of hot liquid into the Reactor Coolant Drain Tank.

b. Waste Holdup Tank

An atmospheric tank is provided to collect equipment drains, valve and pump seal leakoffs (outside the Containment), boron recycle holdup tank overflows, and other water from tritiated aerated sources. The design bases for the Waste Holdup Tank are to provide sufficient surge capacity (3000 gal) to warrant evaporator startup and to provide an additional margin to accept a 10-gpm leak from one unit for eight hours (4800 gal).

c. Waste Evaporator Condensate Tank

There is one Waste Evaporator Condensate Tank, it has a diaphragm to exclude air. This tank collects condensate from the waste evaporator. The tank size is based on providing sufficient condensate tankage to allow a 15-gpm evaporator to operate without interruption for a four-hr period.

d. Floor Drain Tanks

There are three floor drain tanks. These atmospheric tanks are used to collect floor drains from the controlled areas of the primary plant. The tanks provide sufficient surge capacity for the floor drains within their collection areas and in connection with the Waste Holdup Tank, provide surge capacity for abnormal primary system leaks. The design bases for the tanks are the same as for the Waste Holdup Tank.

e. Waste Monitor Tanks

There are two Waste Monitor Tanks. These atmospheric waste monitor tanks are provided to collect floor drain evaporator condensate and/or FDS effluents. The tanks are used for liquid holdup and monitoring prior to reprocessing or discharge from the plant.

f. Laundry and Hot Shower Tank

There is one Laundry and Hot Shower Tank. This atmospheric tank is used to collect hot shower drains within the controlled areas. The tank is sized to furnish a 15-day surge capacity for a twin-unit station during normal operation of both units and a four-day surge capacity during refueling of a single unit.

g. Laundry Holdup and Monitor Tank

There are two Laundry Holdup and Monitor Tanks. These atmospheric tanks are provided to collect processed fluid for sampling and analysis prior to recycle, reprocessing or discharge from the plant.

h. Reverse Osmosis Concentrate Tank

This tank has been abandoned in place. The Reverse Osmosis System will not be used.

i. Not Used

j. Laundry Water Head Tank

The tanks normal discharge has been capped, it can be drained to the Laundry and Hot Shower Tank. Water is fed to this tank from demineralized water storage.

k. Chemical Drain Tank

One tank is provided to collect chemically contaminated and radioactive water from the laboratories. This tankage is sufficient for a two-unit station.

l. Waste Conditioning Tank

One tank is provided to collect evaporator concentrates and chemical drain tank effluents before they are packaged for disposal. By monitoring flow to the tank, the activity and chemistry of the burial packages can be controlled.

m. Spent Resin Storage Tank

The purpose of the Spent Resin Storage Tank is to provide a collection point for spent resin and to allow for decay of short-lived radionuclides before disposal. The tank also serves as a head tank for the Spent Resin Sluice Pump.

One vertical cylindrical tank with sufficient capacity to handle the spent resin storage needs is provided. A vertical cylindrical tank is used because the symmetrical bottom facilitates the removal of resin. The tank is designed so that sufficient pressure can be applied in the gas space of the tank to transfer the resin slurry to the waste processing area.

The spent resin storage tank and associated equipment which can contain radioactive material are shielded to limit the dose to personnel.

n. Evaporator Reagent Tanks

There are two evaporator reagent tanks, one tank is provided with the waste evaporator and the other with the floor drain evaporator. The tanks are used to feed chemicals to their respective evaporators.

o. Plant Effluent Holdup and Monitor Tanks

There are two Plant Effluent Holdup and Monitor Tanks. The tanks are atmospheric waste monitor tanks that collect filter/demineralizer effluent. The tanks are used for liquid holdup and monitoring prior to reprocessing or discharge from the plant. The capacity of each tank is greater than the daily design flow rate so as to provide adequate holding time for sampling and analysis.

3. Demineralizers

The demineralizers are sized for a 35-gpm flow rate and hold 30 ft^3 of ion exchange resin.

a. Waste Evaporator Condensate Demineralizer

One demineralizer is provided to remove trace ionic contaminants from the influent.

b. Waste Monitor Tank Demineralizer

One demineralizer provided to remove trace ionic contaminants from the influent.

4. Filters

The following filters are provided in the LWPS:

a. Waste Evaporator Feed Filter

b. Laundry and Hot Shower Tank Filter

c. Floor Drain Tank Filter

d. Spent Resin Sluice Filter

e. Waste Evaporator Condensate Filter

f. Waste Monitor Tank Filter

The filters are of the disposable-cartridge type and are designed to pass 35 gpm of fluid at a minimal pressure drop. The spent resin sluice filter is of similar design to the other filters, but has a design flow rate of 250 gpm. All but the Laundry and Hot Shower Tank Filter require shielding.

The methods used to change filter elements are dependent on activity levels. If the radiation level of the filter is low enough, the filter elements may be changed manually. If activity levels do not permit manual change, the spent element cartridge assembly is removed remotely with temporary shielding to protect personnel. The spent cartridge assembly is placed in a shielded container for removal to the solid waste disposal area. Filter elements are normally changed because of high differential pressure or high radiation levels. Filter cartridge assembly handling operations are further described in Section 11.4.

5. Strainers

Strainers are provided in the LWPS for the following:

a. Laundry and Hot Shower Tank Strainer

b. Floor Drain Tank strainers

The Laundry and Hot Shower and Floor Drain Tank Strainers are rated at 35 gpm.

Strainers are of the basket type with mesh construction.

The Laundry and Hot Shower and Floor Drain Tank Strainers prevent clogging of filters and lines downstream because of large particles being sluiced through the lines during liquid transfer operations. The basket type laundry and hot shower strainer is not replaced after use but is cleaned and put back into service.

6. Waste and Floor Drain Evaporators

Two 15-gpm forced-recirculation-type evaporators are provided in the LWPS. Evaporator distillate is either recycled, reprocessed, or discharged, depending on its chemistry, radioactivity and operational needs.

The Waste and Floor Drain Evaporators are identical units, and their feed and discharge headers are interconnected so that the Waste and Floor Drain Evaporators or Waste and Boron Recycle Evaporators can be used in place of each other. All evaporators use auxiliary steam from onsite sources as their process heat source. Component cooling water is furnished to the coolers and condensers of the evaporators.

The Waste and Floor Drain Evaporators are maintained as backup equipment for liquid waste processing.

7. Reactor Coolant Drain Tank Heat Exchanger

The Reactor Coolant Drain Tank Heat Exchanger is a U-tube type with one shell pass. The unit is sized to maintain the Reactor Coolant Drain Tank fluid at 170°F or less with a 10-gpm inleakage of reactor coolant at 600°F. The normal temperature is 130°F.

The Reactor Coolant Drain Tank Heat Exchanger can also maintain the Reactor Coolant Drain Tank fluid at 170°F or less with a 25-gpm flow from the Excess Letdown Heat Exchanger during heatup or draining operations and cool the contents of the Pressurizer Relief Tank from 200°F to 120°F in less than eight hours.

8. Reverse Osmosis System

The Reverse Osmosis System will not be utilized as part of the Liquid Waste Processing System. Since an outside vendor is to be utilized to process contaminated plant laundry, the capability to process liquids with high detergent concentrations is not needed. The Reverse Osmosis System has been removed with Reverse Osmosis Concentrates Tank and Reverse Osmosis Concentrates Tank Pump abandoned in place.

9. Tank Overflow Protection

All tanks in the Chemical and Volume Control System (CVCS), Boron Recycle System (BRS), Steam Generator Blowdown Processing System (SGBPS) and WPS that could potentially contain radioactive liquids are designed to provide adequate warning of potential overflow conditions. A summary of the overflow protection features is given in Table 9.3-5. These tanks are provided with level indication instrumentation which has an alarm function on high liquid level in the tank. Alarm annunciation is provided separately on the local system control panel and further relayed to a common annunciator on the main control board in the control room for each system. A description of the level instrumentation provided for these systems is given in Sections 9.3.4.1, 9.3.4.2, Table 9.3-5 and 11.2.2.5 for the CVCS, BRS, SGBPS, and LWPS, respectively.

In addition to tank level monitoring and warning of potential overflow conditions, provisions are made in the systems design to collect and process overflows from tanks containing potentially radioactive liquids.

10. Filter Demineralizer System (FDS)

The Liquid Waste Processing System is equipped with connection to allow for processing of liquid wastes from the following sources through a filter demineralizer system (FDS):

Waste Holdup Tank	Waste Monitor Tanks 1 & 2
Floor Drain Tanks 1, 2, & 3	Laundry & Holdup Monitor
Chemical Drain Tank	Tanks 1 & 2
Laundry & Hot Shower Tank	Radwaste Liner dewatering waste (HIC)
Plant Effluent Holdup and Monitor Tanks 1&2	Recycle Holdup Tanks 1 & 2

Additionally other waste streams may be processed, as necessary, by hose connection to the FDS.

The FDS is a skid mounted system which consists of a number of components which may include:

a. Demineralizer Vessels

b. Booster Pumps

c. Chemical Treatment Systems

d. System Filters

Examples of processing media that may be loaded in the FDS Demineralizer vessels include; activated carbon, organic resin, and/or ion specific media.

Liquid wastes are pumped to the FDS process components. A system booster pump may be utilized to improve system flow rates. The demineralizer vessels may be valved into service or bypassed and sequenced in any order to provide an acceptable cleanup of the process waste stream.

The number and sequence of demineralizer vessels used will depend on vessel media loading and the chemical and radiochemical composition of the waste stream. The vessel configuration and the processing media will be selected to optimize cleanup of waste stream and to minimize solid waste production.

11. Drum Dryer System

Non-routine liquid radioactive water resulting from activities like tank cleaning, facility/equipment decontamination, or tank bottoms that may not be suited for Filter Demineralizer System or floor drain processing may be preconditioned or volume reduced by using a self-contained, skid-mounted drum dryer system located in the Fuel Building. A vacuum is established in the drum as the liquid is heated. Steam from the

drum is condensed. Liquid effluents are routed to floor drains and the remaining solids are handled as Dry Active Waste or sludge.

11.2.2.5 Instrumentation Design

11.2.2.5.1 General

The instrumentation readout is located mainly on the LWPS panel in the Auxiliary Building. Some instruments are read locally.

Alarms are shown separately on the LWPS panel and further relayed to the common LWPS annunciator on the main control board in the Control Room.

Pumps are protected against loss of suction pressure by a control set point on the level instrumentation for the respective vessels feeding the pumps. The Reactor Coolant Drain Tank Pumps and the Spent Resin Sluice Pump are, in addition, interlocked with flow-rate instrumentation and stop operating when the delivery flows reach minimum set points.

Pressure indicators upstream from and downstream of filters, strainers, and demineralizers provide local indications of pressure drops across each component.

Releases to the environment are monitored for radioactivity by radiation detectors. This instrumentation is further described in Section 11.5.

The system instrumentation for Figures 11.2-2 to 11.2-7 is described in detail in Table 11.2-5.

11.2.2.5.2 Tank Overflow Protection

All tanks in the LWPS that could potentially contain radioactive liquids are designed to provide adequate warning of potential overflow conditions.

These tanks are provided with level indication instrumentation which has an alarm function on high liquid level in the tank. Alarm annunciation is provided separately on the local system control panel and further relayed to a common annunciator on the main control board in the Control Room for each system.

In addition to tank level monitoring and warning of potential overflow conditions, provisions are made in the system design to collect and process overflows from tanks containing potentially radioactive liquids.

The Spent Resin Storage Tank (SRST) level instrumentation is interlocked with the SRST vent line isolation valve. On a high level signal the vent line isolation valve will close.

11.2.2.6 Operating Procedure

The LWPS is manually operated in a batch-type method except for some functions of the Reactor Coolant Drain Tank circuit. The system includes adequate control equipment to protect system components and adequate instrumentation and alarm functions to provide operator information to ensure proper system operation. Pumps in the system have low-level shutoffs, and filters and demineralizers have pressure indication upstream and downstream to signify fouling.

1. Normal Operation

Operation of the LWPS is essentially the same during all phases of normal reactor plant operation; the only differences are in the load on the system. The LWPS is not regarded as an Engineered Safety Features (ESF) System.

2. Reactor Coolant Drain Tank Subsystem Operation

Normal operation of the Reactor Coolant Drain Tank is automatic and requires no operator action. The system can be put in the manual mode if desired. The leakage rate into the tank can be estimated by putting the system in the manual mode, stopping the pump, and watching the rate of level change. The venting system is aligned to the Gaseous Waste Processing System when venting is required.

Operation of the system during refueling is the same as for power operation, although the load on the system can be increased when refueling is complete; the water remaining in the canal following normal drain down is pumped to the discharge of the Refueling Water Purification Pump by the Reactor Coolant Drain Tank Pumps. For this operation, the level control valve is opened and the minimum flow line is valved closed.

3. Drain Channel A Subsystem Operation

Water is accumulated in the Waste Holdup Tank until a sufficient quantity exists to warrant waste processing. If the Waste Evaporator is used the distillate may be passed through the Waste Evaporator Condensate Demineralizer before being transferred to the Waste Condensate Tank or the Waste Monitor or Laundry Holdup Monitor Tanks.

The bottoms from the waste evaporator are concentrated to approximately 4 to 5 percent by weight boric acid and are normally recycled to the boric acid tanks. A reagent tank is included for chemical addition to the evaporator in the event of evaporator tube fouling. The Waste Holdup Tank may also be processed through the vendor supplied FDS with the effluent directed to the Waste Monitor Tanks, Laundry Holdup Monitor Tanks, or Plant Effluent Holdup and Monitor Tanks.

Drain Channel A has been provided with interconnections to Drain Channel B as follows:

a. Waste Evaporator inlet to Floor Drain Evaporator inlet

b. Waste Evaporator Condensate Demineralizer outlet to Waste Monitor Tank

c. Floor Drain Tank Filter outlet to Waste Holdup Tank

These interconnections add flexibility to the LWPS and are provided with double valves and telltale leak-detection connections.

4. Drain Channel B Subsystem Operation

Water is accumulated in the plant's three floor drain tanks. Normally, when Floor Drain Tank 1 or 2 is full it is transferred to Floor Drain Tank 3. When Floor Drain Tank 3 is full, the tank may be sampled prior to processing. Floor Drain Tank 3 is normally processed

through the FDS or the Floor Drain Evaporator. The processed water is then subsequently transferred to the Waste Monitor Tanks, Laundry Holdup Monitor Tanks, or Plant Effluent Holdup and Monitor Tanks. The contents of the monitor tanks are then sampled for acceptability to discharge or recycle.

Water leaving Drain Channel B that is found unacceptable for either discharge from the plant or return to the secondary cycle is reprocessed. The monitor tanks are normally discharged into the circulating water tunnel.

The discharge pipeline to the Circulating Water System is common to both Drain Channel B and Channel C and uses a flowmeter, a radiation monitor, and an automatic isolation valve.

Drain Channel B is provided with interconnections to Drain Channel C as follows:

a. Floor Drain Tank 3 Pump discharge to the FDS

b. Laundry and Hot Shower Tank Filter outlet to the Floor Drain Waste Monitor tank inlet header.

A connection from Floor Drain Tanks 1, 2, and 3 to the Waste Conditioning Tank is also provided for the rare instance when fluid in these tanks would have to be disposed of without processing.

5. Drain Channel C Subsystem Operation

Laundry and hot shower water enters the Laundry and Hot Shower Tank for holdup. When the Laundry and Hot Shower Tank is approximately half full, the water is routed via the Laundry and Hot Shower Tank Strainer and Filter to the Waste Monitor Tank and is discharged or reprocessed as needed.

The Laundry Water Head Tank is provided with high- and low-level alarms and remote level indication. Makeup to the tank can be provided from the demineralized water supply system on low level in the tank.

6. Spent Resin Handling Subsystem

This portion of the system sluices resin from the demineralizers, transports resin from the Spent Resin Storage Tank to the waste processing area, and flushes the line from the Waste Evaporator to the waste processing area.

a. Resin Sluicing

Before resin sluicing begins, the demineralizers are backflushed utilizing a closed recirculation flow loop. This loop uses the resin sluice pump or demineralized water line as a motive force and is between the demineralizer vessel and the spent resin storage tank. Resin transfer is then performed by opening the resin sluice outlet and closing the backflush outlet.

Fresh resin is added via the resin fill line and then subsequently the vessel is backflushed to the Spent Resin Storage Tank which removes the resin fines. The valves are then aligned for standby operation.

b. Resin Disposal

Resin may be disposed of by use of a vendor-supplied mobile system via the bulk disposal connection. When sufficient resin has accumulated to warrant disposal, the Spent Resin Storage Tank is pressurized with nitrogen, and resin is transferred to the bulk disposal connection. Upon completion of transfer, the Spent Resin Storage Tank is vented to the plant vent, and flush water is pumped through all lines to ensure resin removal is complete.

7. Potential Bypass Routes

The potential bypass routes in the LWPS are as follows:

a. Drain Channel A Subsystem

Waste accumulated in the waste holdup tank may be evaporated. The evaporator distillate can go directly to the waste evaporator condensate tank, bypassing the waste evaporator condensate demineralizer, unless waste holdup tank samples indicate a high radioactivity level. The Waste Holdup Tank may also be processed through the FDS by bypassing the evaporator purification train.

b. Drain Channel B Subsystem

The wastewater in the Floor Drain Tanks may be sampled to determine the degree of processing required. Based on the result, the waste can be sent directly to a Waste Monitor Tank, Laundry Holdup and Monitor Tank or Plant Effluent Holdup and Monitor Tank, bypassing the Waste Evaporator, the Waste Monitor Tank Demineralizer, and/or the FDS. The Floor Drain Tanks may be processed through the FDS by bypassing the Floor Drain Evaporator Purification train.

11.2.2.7 Faults Of Moderate Frequency

The system is designed to handle the occurrences of equipment faults of moderate frequency such as:

1. Malfunction in the Liquid Waste Processing System

Malfunction in this system can include pump or valve failures or evaporator failure. There is sufficient surge capacity in the system to accommodate waste until the failures can be remedied and normal plant operation resumed.

2. Excessive Leakage in Reactor Coolant System Equipment

The LWPS is designed to handle a 1-gpm reactor coolant leak in addition to the expected leakage during normal operation. Operation of the system is almost the same as for normal operation except the load on the system is increased. A 1-gpm leak into the

Reactor Coolant Drain Tank is handled automatically but may increase the load factor of the recycle evaporator. If a 1-gpm leak enters the Waste Holdup Tank, operation is the same as normal except for the increased load on the evaporator or FDS. Abnormal liquid volumes of reactor coolant or Auxiliary Building equipment leakage can also be accommodated by the Floor Drain Tank and processed by the LWPS.

3. Excessive Leakage in Auxiliary System Equipment

Leakage of this type can include water from steamside leaks and fan cooler leaks inside the Containment which are collected in the Containment sump and sent to a floor drain tank. Other sources can be Component Cooling Water leaks, Service Water leaks, and secondary side water leaks. This water enters the Floor Drain Tank and is processed and discharged as during normal operation.

11.2.3 RADIOACTIVE RELEASES

11.2.3.1 Expected Releases

Expected release quantities in Table 11.2-2c are based on normal operation, including anticipated operational occurrences of the plant and the LWPS, and a realistic estimation of the potential input sources, based on operating experience. The quantities and isotopic concentrations in the liquid discharge from the LWPS are highly dependent upon the actual operation of the plant.

The input sources assumed in the release determinations are summarized in Table 11.2-1. The specific activities used for these computations are detailed in Section 11.1. It is assumed that the waste entering the floor drain tank is 1300 gpd of reactor coolant and 11,700 gpd of non-reactor grade water. The isotopic composition of reactor grade water is as indicated in Section 11.1. The isotopic concentrations at key locations in the LWPS are given in Table 11.2-2. A comparison between the expected release and the limiting concentrations (in accordance with 10 CFR Part 20) are shown in Table 11.2-9.

The total releases, excluding tritium, from the LWPS are approximately 3.3 Ci/yr. The associated release of tritium is less than 2600 Ci/yr. The expected tritium concentration and its influence on plant water balance is discussed in Section 11.2.1.4 of WCAP-8665.

An excessive reactor coolant leakage of 1 gpm in addition to expected leakage during normal operation can be handled with an increased load on the processing equipment. The releases are approximately the same as during normal operation. Therefore, the design objectives outlined in Subsection 11.2.1.1 are met.

The radioactive releases associated with 1 percent failed fuel are given in Table 11.2-10. The releases determined in accordance with NRC Regulatory Guide 1.112 are given in Table 11.2-9.

11.2.3.2 Release Points

Treated liquid effluent is discharged from the LWPS to the environment via the circulating water discharge canal. The release point is indicated on LWPS flow diagrams in the Flow Diagram Volume.

11.2.3.3 Dilution Factors

Treated liquid effluent is diluted by the circulating water flow prior to discharge to ensure that the concentration of radionuclides at the discharge is well below applicable limits. Details on dilution factors are given in Appendix 11A.

11.2.3.4 Estimated Doses

The estimated doses resulting from anticipated annual releases into the Squaw Creek Reservoir are presented in Appendix 11A. The dose estimates are considered historical and are not subject to future updating. The information is retained to avoid loss of original design basis. Actual radioactivity release quantities can be found in the annual radioactive effluent release reports submitted to the NRC.

TABLE 11.2-1

PARAMETERS USED IN THE CALCULATION OF EXPECTED ACTIVITY IN LIQUID WASTES[a] - ORIGINAL DESIGN BASIS[b]

Collector Tank with Sources	Expected Volume of Liquid Waste	Basis	Collection Period Assumed Before Processing
Floor Drain Tank (Dirty Wastes)	13,000 gpd	Derived from 1991 operating data (2.37×10^6 gal/yr/unit)	1.85 days
Waste Holdup Tank (Clean Wastes)	830 gpd	Derived from 1991 operating data (1.51×10^5 gal/yr/unit)	9.64 days
Recycle Holdup Tank (Chemical Shim)	4200 gpd	Derived from 1991 operating data (7.66×10^5 gal/yr/unit)	23.8 days

a) This table does not represent system capacity. The volume of liquid waste assumes two unit operation.

b) Historical, not subject to future updating. Has been retained to preserve original design basis.

TABLE 11.2-2

EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 1 of 9)

a. Tank Inventories

Nuclide	Reactor Coolant Activity (uCi/ml)	Tank Inventory[b] – Ci				
		Floor Drain Tank III	Waste Holdup Tank	Recycle Holdup Tank	Waste Monitor Tank	Plant Effluent Holdup and Monitor Tank
Na-24	4.7E-02	5.3E-01	1.5E+00	2.1E+01	1.8E-04	1.1E-03
Cr-51	3.1E-03	3.5E-02	1.0E-01	1.4E+00	6.5E-05	3.9E-04
Mn-54	1.6E-03	1.8E-02	5.1E-02	7.2E-01	3.5E-05	2.2E-04
Fe-55	1.2E-03	1.4E-02	3.9E-02	5.4E-01	2.6E-05	1.6E-05
Fe-59	3.0E-04	3.4E-03	9.7E-03	1.4E-01	6.5E-06	3.9E-05
Co-58	4.6E-03	5.2E-02	1.5E-01	2.1E+00	1.0E-04	6.0E-04
Co-60	5.3E-04	6.0E-03	1.7E-02	2.4E-01	1.2E-05	7.2E-05
Zn-65	5.1E-04	5.8E-03	1.6E-02	2.3E-01	1.1E-05	6.7E-05
W-187	2.5E-03	2.8E-02	8.0E-02	1.1E+00	1.7E-05	1.0E-04
Np-239	2.2E-03	2.5E-02	7.1E-02	9.9E-01	2.6E-05	1.6E-04
Sr-89	1.4E-04	1.6E-03	4.5E-03	6.3E-02	3.1E-06	1.8E-05
Sr-90	2.0E-05	2.3E-04	6.5E-04	9.0E-03	4.3E-07	2.7E-06
Sr-91	9.6E-04	1.1E-02	3.1E-02	4.3E-01	1.8E-06	1.1E-05
Y-91m	4.6E-04	5.2E-03	1.5E-02	2.1E-01	1.2E-06	7.5E-06

TABLE 11.2-2

EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 2 of 9)

Nuclide	Reactor Coolant Activity (uCi/ml)	Tank Inventory[b] – Ci				
		Floor Drain Tank III	Waste Holdup Tank	Recycle Holdup Tank	Waste Monitor Tank	Plant Effluent Holdup and Monitor Tank
Y-91	5.2E-06	5.9E-05	1.7E-04	2.3E-03	2.5E-07	1.5E-06
Y-93	4.2E-03	4.8E-02	1.4E-01	1.9E+00	9.2E-06	5.6E-05
Zr-95	3.9E-04	4.4E-03	1.3E-02	1.8E-01	8.5E-06	5.1E-05
Nb-95	2.8E-04	3.2E-03	9.0E-03	1.3E-01	6.3E-06	3.7E-05
Mo-99	6.4E-03	7.3E-02	2.1E-01	2.9E+00	8.3E-05	5.4E-04
Tc-99m	4.7E-03	5.3E-02	1.5E-01	2.1E+00	7.8E-05	4.7E-04
Ru-103	8.2E-03	9.3E-02	2.6E-01	3.7E+00	1.6E-04	1.0E-03
Ru-106	1.7E-01	1.8E+00	5.4E+00	7.7E+01	3.7E-04	2.2E-03
Ag-110m	1.3E-03	1.5E-02	4.2E-02	5.8E-01	2.9E-05	1.7E-04
Te-129m	1.9E-04	2.2E-03	6.1E-03	8.5E-02	4.0E-06	2.4E-05
Te-129	2.4E-02	2.7E-01	7.7E-01	1.1E+01	2.6E-06	1.6E-05
Te-131m	1.5E-03	1.7E-02	4.8E-02	6.7E-01	1.2E-05	7.2E-05
Te-131	7.7E-03	8.7E-02	2.5E-01	3.5E+00	2.2E-06	1.2E-05
I-131	4.5E-02	5.1E-01	1.5E+00	2.0E+01	8.5E-05	4.9E-04
Te-132	1.7E-03	1.9E-02	5.5E-02	7.6E-01	2.3E-05	1.4E-04
I-132	2.1E-01	2.4E+00	6.8E+00	9.4E+01	2.3E-05	1.4E-04

TABLE 11.2-2

EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 3 of 9)

Nuclide	Reactor Coolant Activity (uCi/ml)	Tank Inventory[b] – Ci				
		Floor Drain Tank III	Waste Holdup Tank	Recycle Holdup Tank	Waste Monitor Tank	Plant Effluent Holdup and Monitor Tank
I-133	1.4E-01	1.6E+00	4.5E+00	6.2E+01	8.1E-05	4.9E-04
I-134	3.4E-01	3.9E+00	1.1E+01	1.5E+02	4.6E-08	2.7E-07
Cs-134	2.1E-02	2.4E-01	6.8E-01	9.4E+00	5.0E-06	2.9E-05
I-135	2.6E-01	3.0E+00	8.4E+00	1.2E+02	2.6E-05	1.6E-04
Cs-136	8.7E-04	9.9E-03	2.8E-02	3.9E-01	1.8E-07	1.1E-06
Cs-137	1.8E-02	2.1E-01	5.7E-01	8.0E+00	4.4E-06	2.7E-05
Ba-140	1.3E-02	1.5E-01	4.2E-01	5.8E+00	2.5E-04	1.5E-03
La-140	2.5E-02	2.8E-01	8.0E-01	1.1E+01	3.8E-04	2.3E-03
Ce-141	1.5E-04	1.7E-03	4.8E-03	6.7E-02	3.1E-06	1.8E-05
Ce-143	2.8E-03	3.2E-02	9.0E-02	1.3E+00	2.5E-05	1.5E-04
Ce-144	4.4E-03	5.0E-02	1.5E-01	2.0E+00	9.7E-05	5.8E-04
H-3	2.0E+00	2.2E-01	6.5E+01	9.1E+02	4.1E+00	2.2E+01

TABLE 11.2-2
EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 4 of 9)

b. Tank Vents Isotopic Discharge Rate (uCi/min)

Vent No.	Vent Description	Temperature (F)	Pressure (psig)	Flow Rate (cc/min)	Kr-85m	Kr-85
1	Reactor coolant drain tank to GWPS	130	5	8.8	1.07E-01	1.76E+000
2	Waste holdup tank vent to plant vent	amb	atm	431	neg	neg
3	Evaporator package vents to plant vent	150	2	6.0	neg	neg
4	Floor drain tank vent to plant vent	amb	atm	2203	neg	neg
5	Waste monitor tanks local vent	amb	atm	2148	neg	neg
6	Chemical drain tank to plant vent	amb	atm	7.56	neg	neg
7	Laundry and hot shower local vent	Var	atm	866	neg	neg
8	Laundry holdup and monitor tank local vent	amb	atm	866	neg	neg
9	Plant Effluent Holdup and Monitor Tanks[c]					

TABLE 11.2-2
EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 5 of 9)

Vent No.	Kr-87	Kr-88	XE-133m	XE-133	XE-135	I-131	I-133
1	1.76E-02	1.22E-01	1.21E+00	1.31E+02	6.11E-01	2.38E-04	2.64E-04
2	neg	neg	neg	neg	neg	1.77E-03	3.28E-04
3	neg	neg	neg	neg	neg	neg	neg
4	neg	neg	neg	neg	neg	1.56E-03	5.95E-04
5	neg	neg	neg	neg	neg	1.55E-07	5.8E-08
6	neg	neg	neg	neg	neg	4.08E-06	6.2E-07
7	neg	neg	neg	neg	neg	neg	neg
8	neg	neg	neg	neg	neg	neg	neg
9[c]							

TABLE 11.2-2
EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 6 of 9)

c. Processing Lines

Nuclide	Reactor Coolant Activity (uCi/ml)	Activity in Line[d] – uCi/ml					
		Line No. 1	Line No. 2	Line No. 3	Line No. 4	Line No. 5	Line No. 6
Na-24	4.7E-02	4.7E-03	4.7E-05	4.0E-02	4.0E-04	4.5E-02	8.9E-10
Cr-51	3.1E-03	3.1E-04	3.1E-06	2.6E-03	2.6E-05	2.9E-03	5.9E-11
Mn-54	1.6E-03	1.6E-04	1.6E-06	1.4E-03	1.4E-05	1.5E-03	3.0E-11
Fe-55	1.2E-03	1.2E-04	1.2E-06	1.0E-03	1.0E-05	1.1E-03	2.3E-11
Fe-59	3.0E-04	3.0E-05	3.0E-07	2.6E-04	2.6E-06	2.9E-04	5.7E-12
Co-58	4.6E-03	4.6E-04	4.6E-06	3.9E-03	3.9E-05	4.4E-03	8.7E-11
Co-60	5.3E-04	5.3E-05	5.3E-07	4.5E-04	4.5E-06	5.0E-04	1.0E-11
Zn-65	5.1E-04	5.1E-05	5.1E-07	4.3E-04	4.3E-06	4.8E-04	9.7E-12
W-187	2.5E-03	2.5E-04	2.5E-06	2.1E-03	2.1E-05	2.4E-03	4.8E-11
Np-239	2.2E-03	2.2E-04	2.2E-06	1.9E-03	1.9E-05	2.1E-03	4.2E-11
Sr-89	1.4E-04	1.4E-05	1.4E-07	1.2E-04	1.3E-06	1.3E-04	2.7E-12
Sr-90	2.0E-05	2.0E-06	2.0E-08	1.7E-05	1.7E-07	1.8E-05	3.8E-13
Sr-91	9.6E-04	9.6E-05	9.6E-07	8.2E-04	8.2E-06	9.1E-04	1.8E-11
Y-91m	4.6E-04	4.6E-05	4.6E-07	3.9E-04	3.9E-06	4.4E-04	8.7E-12
Y-91	5.2E-06	5.2E-07	5.2E-09	4.4E-06	4.4E-08	4.9E-06	9.9E-14

TABLE 11.2-2
EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 7 of 9)

Nuclide	Reactor Coolant Activity (uCi/ml)	Activity in Line[d] – uCi/ml					
		Line No. 1	Line No. 2	Line No. 3	Line No. 4	Line No. 5	Line No. 6
Y-93	4.2E-03	4.2E-04	4.2E-06	3.6E-03	3.6E-05	4.0E-03	8.0E-11
Zr-95	3.9E-04	3.9E-05	3.9E-07	3.3E-04	3.3E-06	3.7E-04	7.4E-12
Nb-95	2.8E-04	2.8E-05	2.8E-07	2.4E-04	2.4E-06	2.7E-04	5.3E-12
Mo-99	6.4E-03	6.4E-04	6.4E-06	5.4E-03	5.4E-05	6.1E-03	1.2E-10
Tc-99m	4.7E-03	4.7E-04	4.7E-06	4.0E-03	4.0E-05	4.5E-03	8.9E-11
Ru-103	8.2E-03	8.2E-04	8.2E-06	7.0E-03	7.0E-05	7.7E-03	1.5E-10
Ru-106	1.7E-01	1.7E-02	1.7E-04	1.4E-01	1.4E-03	1.6E-01	3.2E-09
Ag-110m	1.3E-03	1.3E-04	1.3E-06	1.1E-03	1.1E-05	1.2E-03	2.5E-11
Te-129m	1.9E-04	1.9E-05	1.9E-07	1.6E-04	1.6E-06	1.8E-04	3.6E-12
Te-129	2.4E-02	2.4E-03	2.4E-05	2.0E-02	2.0E-04	2.3E-02	4.6E-10
Te-131m	1.5E-03	1.5E-04	1.5E-06	1.3E-03	1.3E-05	1.4E-03	2.9E-11
Te-131	7.7E-03	7.7E-04	7.7E-06	6.5E-03	6.5E-05	7.3E-03	1.5E-10
I-131	4.5E-02	4.5E-03	4.5E-06	3.8E-02	3.8E-05	4.3E-02	8.6E-09
Te-132	1.7E-03	1.7E-04	1.7E-06	1.4E-03	1.4E-05	1.6E-03	3.2E-11
I-132	2.1E-01	2.1E-02	2.1E-05	1.8E-01	1.8E-04	2.0E-01	4.0E-08
I-133	1.4E-01	1.4E-02	1.4E-05	1.2E-01	1.2E-04	1.3E-01	2.7E-08

TABLE 11.2-2
EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 8 of 9)

Nuclide	Reactor Coolant Activity (uCi/ml)	Activity in Line[d] – uCi/ml					
		Line No. 1	Line No. 2	Line No. 3	Line No. 4	Line No. 5	Line No. 6
I-134	3.4E-01	3.4E-02	3.4E-05	2.9E-01	2.9E-04	3.2E-01	6.5E-08
Cs-134	2.1E-02	2.1E-03	2.1E-07	1.8E-02	1.8E-06	2.0E-02	3.8E-10
I-135	2.6E-01	2.6E-02	2.6E-05	2.2E-01	2.2E-04	2.5E-01	4.9E-08
Cs-136	8.7E-04	8.7E-05	8.7E-09	7.4E-04	7.4E-08	8.3E-04	1.7E-11
Cs-137	1.8E-02	1.8E-03	1.8E-07	1.5E-02	1.5E-06	1.7E-02	3.4E-10
Ba-140	1.3E-02	1.3E-03	1.3E-05	1.1E-02	1.1E-04	1.2E-02	2.5E-10
La-140	2.5E-02	2.5E-03	2.5E-05	2.1E-02	2.1E-04	2.4E-02	4.8E-10
Ce-141	1.5E-04	1.5E-05	1.5E-07	1.3E-04	1.3E-06	1.4E-04	2.9E-12
Ce-143	2.8E-03	2.8E-04	2.8E-06	2.4E-03	2.4E-05	2.7E-03	5.3E-11
Ce-144	4.4E-03	4.4E-04	4.4E-06	3.7E-03	3.7E-05	4.2E-03	8.3E-11
H-3	2.0E+00	2.0E-01	2.0E-01	1.7E+00	1.7E+00	1.9E+00	1.9E+00

TABLE 11.2-2
EXPECTED PROCESS PARAMETERS FOR THE LIQUID WASTE PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

(Sheet 9 of 9)

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a) Historical, not subject to future updating. Has been retained to preserve original design basis.

b) All tanks are at ambient temperature and atmospheric pressure. Tank volume are as follows:

Floor Drain Tank III	30,000 gallons
Waste Holdup Tank	10,000 gallons
Recycle Holdup Tank	112,000 gallons
Waste Monitor Tank	5,000 gallons
Plant Effluent Holdup and Monitor Tank[e]	30,000 gallons

Tank inventories (except for the Waste Monitor Tank) are derived from NUREG-0017 primary coolant activities. The Waste Monitor Tank inventory is derived from NUREG-0017 effluent release concentrations which includes 0.32 Ci/yr (two units) for anticipated operational occurrences.

c) Plant Effluent Holdup and Monitor Tanks serve as an alternate to the Waste Monitor Tanks when they are utilized.

d) Line No. 1 - Output from Floor Drain Tank III to Filter Demineralizer System
Line No. 2 - Treated Floor Drain Tank III fluid output from the Filter Demineralizer System to the Waste Monitor Tank
Line No. 3 - Output from Waste Holdup Tank to Filter Demineralizer System
Line No. 4 - Treated Waste Holdup Tank fluid output from the Filter Demineralizer System to the Waste Monitor Tank
Line No. 5 - Output from Recycle Holdup Tank to Boron Recycle System evaporator
Line No. 6 - Boron Recycle System evaporator output to the Waste Monitor Tank

Processing line concentrations are derived from NUREG-0017 primary coolant activities.

e) Plant Effluent Holdup and Monitor Tanks serve as an alternate to the Waste Monitoring Tanks and therefore process at the same activity concentrations as the Waste Monitor Tanks when they are utilized.

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 1 of 15)

Components	Parameters
Pumps	
1. Reactor Coolant Drain Tank Pumps	
Quantity	Four
Type	Canned
Design pressure, psig	150
Design temperature, F	200
Design Flow, gpm	
1	100
2	140
Design Head, ft	
1	300
2	250
Material	SS
2. Waste Evaporator Feed Pump	
Quantity	One
Type	Canned
Design pressure, psig	150
Design temperature, F	200
Design Flow, gpm	
1	35
2	100
Design Head, ft	
1	250
2	200
Material	SS

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 2 of 15)

Components	Parameters
3. Waste Evaporator Condensate Tank Pump	
Quantity	One
Type	Canned
Design pressure, psig	150
Design temperature, F	200
Design Flow, gpm	
1	35
2	100
Design Head, ft	
1	250
2	200
Material	SS
4. Chemical Drain Tank Pump	
Quantity	One
Type	Canned
Design pressure, psig	150
Design temperature, F	200
Design Flow, gpm	
1	35
2	100
Design Head, ft	
1	250
2	200
Material	SS
5. Spent Resin Sluice Pump	
Quantity	One
Type	Canned

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 3 of 15)

Components	Parameters
Design pressure, psig	150
Design temperature, F	180
Design Flow, gpm	
1	140
Design Head, ft	
1	250
Material	SS
6. Laundry and Hot Shower Tank Pump	
Quantity	One
Type	Mechanical seal
Design pressure, psig	150
Design temperature, F	180
Design Flow, gpm	
1	35
Design Head, ft	
1	250
Material	SS
7. Floor Drain Tank Pumps	
Quantity	Three

	Pump #1 and Pump #2	Pump #3
Type	Mech. Seal Centrifugal	Canned Motor/ Single
Design Pressure, (psig)	150	150
Design Temperature, (F)	200	120
Design Flow, (gpm)	35	5
Design Head, (ft)	250	230
Material	SS	SS

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 4 of 15)

Components	Parameters
8. Waste Monitor Tank Pumps	
Quantity	Two
Type	Canned
Design pressure, psig	150
Design temperature, F	200
Design Flow, gpm	
1	35
2	100
Design Head, ft	
1	250
2	200
Material	SS
9. Reverse Osmosis Concentrates Tank Pump[(a)]	
Quantity	One
Type	Canned
Design pressure, psig	150
Design temperature, F	200
Design Flow, gpm	
1	10
Design Head, ft	
1	115
Material	SS
10. Laundry Holdup and Monitor Tank Pump	
Quantity	Two
Type	Canned
Design pressure, psig	150

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 5 of 15)

Components	Parameters
Design temperature, F	200
Design Flow, gpm	35
Design Head, ft	230
Material	SS
11. Plant Effluent Holdup and Monitor Tank Pumps	
Quantity	Two
Type	Centrifugal
Design pressure, psig	150
Design temperature, F	200
Design Flow, gpm	100
Design Head, ft	231
Material	SS
Heat Exchangers	
1. Reactor Coolant Drain Tank Heat Exchangers	
Quantity	Two
Type	U-tube
Estimated UA, Btu/hr/F	70,000
Design Pressure, psig	
Shell	165
Tube	240
Design Temperature, F	
Shell	250
Tube	200
Design Flow, lb/hr	
Shell	112,000
Tube	44,600

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 6 of 15)

Components	Parameters
Temperature (In), F	
Shell	105
Tube	180
Temperature (Out), F	
Shell	125
Tube	130
Material	
Shell	CS
Tube	SS
Tanks	
1. Reactor Coolant Drain Tanks	
Quantity	Two
Usable volume, gal	350
Type	Horizontal
Design pressure, psig[(b)]	100
Design temperature, F	250
Material	SS
Diaphragm	No
2. Waste Holdup Tank	
Quantity	One
Usable volume, gal	10,000
Type	Vertical
Design pressure, psig	Atmospheric
Design temperature, F	200
Material	SS
Diaphragm	No

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 7 of 15)

Components	Parameters
3. Waste Evaporator Condensate Tanks	
Quantity	One
Usable volume, gal	5000
Type	Vertical
Design pressure, psig	Atmospheric
Design temperature, F	200
Material	SS
Diaphragm	Yes
4. Chemical Drain Tank	
Quantity	One
Usable volume, gal	600
Type	Vertical
Design pressure, psig	Atmospheric
Design temperature, F	200
Material	SS
Diaphragm	No
5. Spent Resin Storage Tank	
Quantity	One
Usable volume, gal	4100
Type	Vertical
Design pressure, psig	150
Design temperature, F	200
Radiation level inside compartment, rads/hr	1000
Material	SS
Diaphragm	No

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 8 of 15)

Components	Parameters	
6. Laundry and Hot Shower Tank		
Quantity	One	
Usable volume, gal	10,000	
Type	Vertical	
Design pressure, psig	Atmospheric	
Design temperature, F	200	
Material	SS	
Diaphragm	No	
7. Floor Drain Tanks	NNS	VIII
Quantity	Three	
Usable volume, gal	10,000 (Two tanks) 30,000 (One tank)	
Type	Vertical	
Design pressure, psig	Atmospheric	
Design temperature, F	200	
Material	SS	
Diaphragm	No	
8. Waste Monitor Tanks		
Quantity	Two	
Usable volume, gal	5000	
Type	Vertical	
Design pressure, psig	Atmospheric	
Design temperature, F	200	
Material	SS	
Diaphragm	No	

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 9 of 15)

Components	Parameters
9. Laundry Holdup and Monitor Tanks	
Quantity	Two
Usable volume, gal	5000
Type	Vertical
Design pressure, psig	Atmospheric
Design temperature, F	150
Material	SS
Diaphragm	No
10. Reverse Osmosis Concentrates Tank[(a)]	
Quantity	One
Usable volume, gal	1000
Type	Vertical
Design pressure, psig	Atmospheric
Design temperature, F	200
Material	SS
Diaphragm	No
11. Laundry Water Head Tank	
Quantity	One
Usable volume, gal	5000
Type	Vertical
Design pressure, psig	Atmospheric
Design temperature, F	200
Material	SS
Diaphragm	No

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 10 of 15)

Components	Parameters
12. Evaporator Reagent Tanks	
Quantity	Two
Usable volume, gal	5
Type	Vertical
Design pressure, psig	150
Design temperature, F	200
Material	SS
Diaphragm	No
13. Plant Effluent Holdup and Monitor Tanks	
Quantity	Two
Usable Volume, gal.	29,000
Type	Vertical
Design Pressure, psig.	Atmospheric
Design Temperature, F.	200
Material	Reinforced Concrete w/ Stainless Steel Liner
Diaphragm	No
Demineralizers	
1. Waste Evaporator Condensate Demineralizer	
Quantity	One
Type	Flushable
Design pressure, psig	300
Design temperature, F	250
Design flow, gpm	35 (Max. 120)
Resin volume, ft^3	30
Material	SS

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 11 of 15)

Components	Parameters
Resin type	IRN-150(c)
Design process decontamination factor	No
2. Waste Monitor Tank Demineralizer	
Quantity	One
Type	Flushable
Design pressure, psig	300
Design temperature, F	250
Design flow, gpm	35 (Max. 120)
Resin volume, ft^3	30
Material	SS
Resin type	IRN-150(c)
Design process decontamination factor	10
Filters	
1. Waste Evaporator Feed Filter	
Quantity	One
Design pressure, psig	200
Design temperature, F	250
Design flow, gpm	35
Micron rating	20 to 100 absolute (100% retention)
Surface radiation level, rads/hr	100
Materials	
Housing	SS
Filter element	EICF(d)

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 12 of 15)

Components	Parameters
2. Waste Evaporator Condensate Filter	
Quantity	One
Design pressure, psig	200
Design temperature, F	250
Design flow, gpm	35
Micron rating	2 to 40 absolute (100% retention)
Surface radiation level, rads/hr	<1
Materials	
Housing	SS
Filter element	EICF or equivalent
3. Spent Resin Sluice Filter	
Quantity	One
Design pressure, psig	300
Design temperature, F	250
Design flow, gpm	250
Micron rating	2 to 40 absolute (100% retention)
Surface radiation level, rads/hr	<100
Materials	
Housing	SS
Filter element	EICF or equivalent
4. Laundry and Hot Shower Tank Filter	
Quantity	One
Design pressure, psig	200
Design temperature, F	250
Design flow, gpm	35

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 13 of 15)

Components	Parameters
Micron rating	10 to 40 absolute (100% retention)
Surface radiation level, mr/hr	<100
Materials	
Housing	SS
Filter element	EICF or equivalent
5. Floor Drain Tank Filter	
Quantity	One
Design pressure, psig	200
Design temperature, F	250
Design flow, gpm	35
Micron rating	2 to 40 absolute (100% retention)
Surface radiation level, rads/hr	100
Materials	
Housing	SS
Filter element	Resin Bonded Glass Fiber or equivalent
6. Waste Monitor Tank Filter	
Quantity	One
Design pressure, psig	200
Design temperature, F	250
Design flow, gpm	35
Micron rating	10 to 40 absolute (100% retention)
Surface radiation level, rads/hr	90
Materials	
Housing	SS
Filter element	EICF or equivalent

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 14 of 15)

Components	Parameters
Strainers	
1. Laundry and Hot Shower Tank Strainer	
Quantity	One
Type	Basket
Design pressure, psig	150
Design temperature, F	200
Design flow, gpm	35
Pressure drop at design flow, psi	0.2 (Clean)
Nominal rating, in.	0.0625
Surface radiation level	Negligible
Material	SS
2. Floor Drain Tank Strainers	
Quantity	Three
Type	Basket
Design pressure, psig	150
Design temperature, F	200
Design flow, gpm	35
Pressure drop at design flow, psi	0.2 (Clean)
Nominal rating, in.	1/16
Surface radiation level	Negligible
Material	SS
Evaporators	
1. Waste Evaporator Process side Steam side	
Quantity	One
Steam design pressure, psig	50

TABLE 11.2-3
EQUIPMENT PRINCIPAL DESIGN PARAMETERS

(Sheet 15 of 15)

Components	Parameters
Design flow, gpm	15
Feed concentration, mg/l boron	10-2500
Bottoms concentration, mg/l boron	7000-21,000
Design process decontamination factor	1000
2. Floor Drain Evaporator Process side Steam side	
Quantity	One
Steam design pressure, psig	50
Design flow, gpm	15
Feed concentration, mg/l boron	10-2500
Bottoms concentration, mg/l boron	7000-21,000
Design process decontamination factor	1000

a) Refer to subsection 11.2.2.4.2 for additional information on the reverse osmosis system.

b) External design pressure is 60 psig.

c) Rohm and Haas Amberlite or equivalent

d) Epoxy-impregnated cellulose fiber or equivalent.

TABLE 11.2-4
LOCATIONS OF LIQUID WASTE PROCESSING SYSTEM EQUIPMENT

(Sheet 1 of 3)

Equipment	Quantity	Building	Elevation
Reactor Coolant Drain Tanks			
Unit 1	1	Containment	808 ft 0 in
Unit 2	1	Containment	808 ft 0 in
Reactor Coolant Drain Tank Pumps			
Unit 1	2	Containment	808 ft 0 in
Unit 2	2	Containment	808 ft 0 in
Reactor Coolant Drain Tank Heat Exchangers			
Unit 1	1	Containment	808 ft 0 in
Unit 2	1	Containment	808 ft 0 in
Spent resin storage tank	1	Auxiliary	810 ft 6 in
Spent resin sluice pump	1	Auxiliary	810 ft 6 in
Spent resin sluice filter	1	Auxiliary	842 ft
Waste holdup tank	1	Auxiliary	790 ft 6 in
Waste evaporator feed pump	1	Auxiliary	790 ft 6 in
Waste evaporator feed filter	1	Auxiliary	842 ft
Waste evaporator	1	Auxiliary	810 ft 6 in
Waste evaporator condensate demineralizer	1	Auxiliary	832 ft 6 in
Waste evaporator condensate filter	1	Auxiliary	842 ft
Waste evaporator condensate tank	1	Auxiliary	790 ft 6 in
Waste evaporator condensate tank pump	1	Auxiliary	790 ft 6 in
Chemical drain tank	1	Auxiliary	790 ft 6 in

TABLE 11.2-4
LOCATIONS OF LIQUID WASTE PROCESSING SYSTEM EQUIPMENT

(Sheet 2 of 3)

Equipment	Quantity	Building	Elevation
Chemical drain tank pump	1	Auxiliary	790 ft 6 in
Floor drain tank I (10,000 gal)	1	Unit 1 safeguards	773 ft 0 in
Floor drain tank II (10,000 gal)	1	Unit 2 safeguards	773 ft 0 in
Floor drain tank III (30,000 gal)	1	Auxiliary	790 ft 6 in
Floor drain tank pump I with strainer	1	Unit 1 safeguards	773 ft 0 in
Floor drain tank pump II with strainer	1	Unit 2 safeguards	773 ft 0 in
Floor drain tank pump III with strainer	1	Auxiliary	790 ft 6 in
Floor drain tank filter	1	Auxiliary	842 ft
Floor drain evaporator	1	Auxiliary	810 ft 6 in
Waste monitor tank demineralizer	1	Auxiliary	832 ft
Waste monitor tank filter	1	Auxiliary	842 ft 6 in
Waste monitor tank	2	Auxiliary	790 ft 6 in
Waste monitor tank pumps	2	Auxiliary	790 ft 6 in
Laundry and hot shower tank	1	Auxiliary	790 ft 6 in
Laundry and hot shower tank strainer	1	Auxiliary	790 ft 6 in
Laundry and hot shower tank filter	1	Auxiliary	790 ft 6 in
Laundry holdup and monitor tanks	2	Auxiliary	790 ft 6 in
Laundry holdup and monitor tank pump	1	Auxiliary	790 ft 6 in
Laundry head tank	1	Auxiliary	852 ft 6 in

TABLE 11.2-4
LOCATIONS OF LIQUID WASTE PROCESSING SYSTEM EQUIPMENT

(Sheet 3 of 3)

Equipment	Quantity	Building	Elevation
Evaporator reagent tanks	2	Auxiliary	810 ft 6 in
Laundry and hot shower tank pump	1	Auxiliary	790 ft 6 in
Waste Conditioning	1	Fuel	841 ft 0 in
Plant effluent holdup and monitor tanks	2	Yard	813 ft 6 in
Plant effluent holdup and monitor tank pumps	2	Fuel	810 ft 6 in
Filter Demineralizer Skid	1	Fuel	800 ft 2 in
Resin Dewatering Skid	1	Fuel	800 ft 2 in
Rad Vaults	1 or 2	Fuel	800 ft 2 in
Drum Dryer System	1	Fuel	(mobile)

TABLE 11.2-5
LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[(a)]

(Sheet 1 of 7)

Channel No.	Location Primary Sensor	Range	Alarm Function	Control Function	Location of Readout
Flow Instrumentation					
X-F-1007	Waste evaporator pump discharge	0 to 30 gpm	N/A	N/A	Local
1/2-F-1008	Reactor coolant drain tank pump discharge	0 to 250 gpm	N/A	Low	LWPS panel
1/2-F-1009	Reactor coolant drain tank recirculation	0 to 100 gpm	Low	N/A	LWPS panel
X-F-1011	Spent resin sluice pump discharge	0 to 100 gpm	Low	Low	LWPS panel
X-F-1085A, C	Waste monitor tank pump No. 1 discharge	0 to 100 gpm	N/A	N/A	LWPS panel and local
X-F-1085B, D	Waste monitor tank pump No. 2 discharge	0 to 100 gpm	N/A	N/A	LWPS panel and local
X-F-5286	Laundry holdup and monitor tank pump discharge	0 to 150 gpm	N/A	N/A	LWPS panel
1/2-Q-1014	Reactor coolant drain tank discharge	N/A	N/A	N/A	Local

TABLE 11.2-5
LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[(a)]

(Sheet 2 of 7)

Channel No.	Location Primary Sensor	Range	Alarm Function	Control Function	Location of Readout
Pressure Instrumentation					
1/2-P-1004	Reactor coolant drain tank	0 to 100 psig	High	N/A	LWPS panel
X-P-1006	Spent resin storage tank	0 to 100 psig	High	N/A	LWPS and drumming panels
X-P-1016	Waste evaporator feed pump discharge	0 to 160 psig	N/A	N/A	Local
X-P-1017	Waste evaporator feed header	0 to 160 psig	N/A	N/A	Local
1/2-P-1018A	Reactor coolant drain tank pump No. 1 discharge	0 to 160 psig	N/A	N/A	Local
1/2-P-1018B	Reactor coolant drain tank pump No. 2 discharge	0 to 160 psig	N/A	N/A	Local
X-P-1018C	Laundry and hot shower tank pump discharge	0 to 150 psig	N/A	N/A	Local
X-P-1018D	Chemical drain tank pump discharge	0 to 150 psig	N/A	N/A	Local
X-P-1018G	Waste evaporator condensate pump discharge	0 to 160 psig	N/A	N/A	Local
X-P-1074	Waste evaporator outlet	0 to 160 psig	N/A	N/A	Local
X-P-1075	Waste evaporator condensate demineralizer outlet	0 to 150 psig	N/A	N/A	Local

TABLE 11.2-5
LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[a]

(Sheet 3 of 7)

Channel No.	Location Primary Sensor	Range	Alarm Function	Control Function	Location of Readout
X-P-1076	Waste evaporator condensate filter	0 to 160 psig	N/A	N/A	Local
X-P-1078	Floor drain tank filter inlet	30" Hg to 150 psig	N/A	N/A	Local
X-P-1079	Floor drain tank filter outlet	0 to 150 psig	N/A	N/A	Local
X-P-1080	Laundry and hot shower tank filter inlet	0 to 150 psig	N/A	N/A	Local
X-P-1081	Laundry and hot shower tank filter outlet	0 to 150 psig	N/A	N/A	Local
1/2-P-1024	Reactor Coolant Drain Tank Vent Pressure	0 to 150 psig	High	N/A	Local
X-P-1084A	Waste monitor tank pump No. 1 discharge	0 to 150 psig	N/A	N/A	Local
X-P-1084B	Waste monitor tank pump No. 2 discharge	0 to 150 psig	N/A	N/A	Local
X-P-5274	Laundry holdup and monitor tank pump discharge	0 to 160 psig	N/A	N/A	Local
X-P-1086	Resin sluice filter inlet	0 to 150 psig	N/A	N/A	Local
X-P-1087	Resin sluice filter outlet	0 to 150 psig	N/A	N/A	Local
X-P-1088	Waste monitor tank filter inlet	0 to 150 psig	N/A	N/A	Local

TABLE 11.2-5
LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[a]

(Sheet 4 of 7)

Channel No.	Location Primary Sensor	Range	Alarm Function	Control Function	Location of Readout
X-P-1089	Waste monitor tank filter outlet				
X-P-1090A	Floor drain tank pump No. 1 discharge	0 to 150 psig	N/A	N/A	Local
X-P-1090B	Floor drain tank pump No. 2 discharge	0 to 150 psig	N/A	N/A	Local
X-P-5279	Floor drain tank pump No. 3 discharge	0 to 150 psig	N/A	N/A	Local
X-P-5272[b]	Reverse osmosis concentrates tank pump discharge				
X-P-5300A	Plant effluent holdup and monitor tank No. 1	0 to 200 psig	N/A	N/A	Local
X-P-5300B	Plant effluent holdup and monitor tank No. 2	0 to 200 psig	N/A	N/A	Local
Level Instrumentation					
X-L-1001	Waste holdup tank	0 to 100%	High-high High Low	Low	Local and LWPS panel
X-L-1002	Chemical drain tank	0 to 100%	High Low	Low	Local and LWPS drumming panels

TABLE 11.2-5
LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[a]

(Sheet 5 of 7)

Channel No.	Location Primary Sensor	Range	Alarm Function	Control Function	Location of Readout
1/2-L-1003	Reactor coolant drain tank	0 to 100%	High Low	Low	LWPS panel
X-L-1005	Spent resin storage tank	0 to 100%	High Low	High Low	LWPS and drumming panels
X-L-1010	Laundry and hot shower tank	0 to 100%	High Low	Low	LWPS panel and local
X-L-1012	Waste evaporator condensate tank	0 to 100%	High Low	Low	LWPS panel and local
X-L-1077A	Floor drain tank 1	0 to 100%	High Low	Low	LWPS panel and local
X-L-1077B	Floor drain tank 2	0 to 100%	High Low	Low	LWPS panel and local
X-L-5278	Floor drain tank 3	0 to 100%	High Low	Low	LWPS panel and local
X-L-1082	Waste monitor tank No. 1	0 to 100%	High Low	Low	LWPS panel and local
X-L-1083	Waste monitor tank No. 2	0 to 100%	High Low	Low	LWPS panel and local
X-L-5275	Laundry holdup and monitor tank No. 1	0 to 100% 0' to 10'	High Low	Low	LWPS panel and local

TABLE 11.2-5
LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[a]

(Sheet 6 of 7)

Channel No.	Location Primary Sensor	Range	Alarm Function	Control Function	Location of Readout
X-L-5276	Laundry holdup and monitor tank No. 2	0 to 100% 0' to 10'	High Low	Low	LWPS panel and local
X-L-5287[c]	Laundry water head tank	0 to 100% 1' to 13'	High Low	Low	LWPS panel and local
X-L-5271A[d]	Reverse osmosis concentrates tank				
X-L-5271B[d]	Reverse osmosis concentrates tank				
X-L-5300A, C, E	Plant effluent holdup and monitor tank No. 1	0 to 100% 0' to 24.5'	High-High Low Low-Low	High Low	Local
X-L-5300B, D, F	Plant effluent holdup and monitor tank No. 2	0 to 100% 0' to 24.5'	High-High Low Low-Low	High Low	Local
Temperature Instrumentation					
1/2-T-1058	Reactor coolant drain tank	250	50 to 250°F	High	LWPS Panel

TABLE 11.2-5
LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[a]

(Sheet 7 of 7)

Channel No.	Location Primary Sensor	Range	Alarm Function	Control Function	Location of Readout
Radiation Instrumentation					
X-RE-5253	Laundry holdup and monitor tank discharge line	1E-05 to 5E-02 µCi/cm^3	High	Variable	LWPS and radiation monitor panels
X-RE-5251A	Common discharge line of AB Sump 3,11, DG Sumps and CCW Drain Tanks	1E-05 to 5E-02 µCi/cm^3	High	Variable	CPX-ECPRLV-01

a) The following abbreviations are used in this table only:

F	=	flow	R	=	radiation
Q	=	flow integrator	X	=	Common instrument
P	=	pressure	1/2	=	Unit 1 and Unit 2 instruments
L	=	level			
T	=	temperature			

b) Refer to subsection 11.2.2.4.2 for additional information on the reverse osmosis system.

c) X-LS-5287A has control function; X-LS-5287 indicates 0 to 100% tank level.

d) Refer to subsection 11.2.2.4.2 for additional information on the reverse osmosis system.

TABLE 11.2-6
DECONTAMINATION FACTORS FOR LIQUID WASTE PROCESSING SYSTEM EQUIPMENT

Process Stream	Anion	Cs,Rb	Other Nuclides
Dirty Wastes	10^3	10^4	10^2
(Filter Demineralizer System[(a)])	(Max.)	(Max.)	(Max.)
Clean Wastes	10^3	10^4	10^2
Filter Demineralizer System[(a)])			

	Iodine	All Nuclides
Chemical Shim (Total)	$5x10^6$	$5x10^7$
CVCS Letdown	10^2	50
Evaporator Feed	10^2	10^2
Evaporator Condensate	5	10
Boron Recycle System	10^2	10^3

a) The Filter Demineralizer System (FDS) has multiple demineralizers of various sizes which may be operated in series or parallel. These vessels may be loaded with any combination of organic resin, charcoals, or ion specific media and may be customized according to the isotopes in the process stream. In addition, the waste monitor tank mixed bed demineralizer was included, but is typically not used. The FDS also has chemical injection for waste stabilization and filtration.

TABLE 11.2-7
THIS TABLE HAS BEEN DELETED.

TABLE 11.2-8
LIQUID WASTE PROCESSING SYSTEM COMPONENT DESIGN INVENTORIES - ORIGINAL LICENSING BASIs[a]

(Sheet 1 of 3)

Nuclide	Reactor Coolant Activity at 1% Fuel Defects (uCi/gm)	Tank Inventory (Ci)			
		Floor Drain Tank III	Waste Holdup Tank	Recycle Holdup Tank	Waste Monitor Tank
Cr-51	5.5E-03	6.2E-02	1.8E-01	2.5E+00	1.2E-04
Mn-54	4.4E-04	5.0E-03	1.4E-02	2.0E-01	9.6E-06
Fe-55	2.0E-03	2.3E-02	6.5E-02	9.0E-01	4.3E-05
Fe-59	5.2E-04	5.9E-03	1.7E-02	2.4E-01	1.1E-05
Co-58	1.5E-02	1.7E-01	4.9E-01	6.8E+00	3.3E-04
Co-60	1.9E-03	2.2E-02	6.1E-02	8.6E-01	4.3E-05
Sr-89	4.3E-03	4.9E-02	1.4E-01	1.9E+00	9.5E-05
Sr-90	2.0E-04	2.3E-03	6.5E-03	9.0E-02	4.3E-06
Sr-91	6.2E-03	7.1E-02	2.0E-01	2.8E+00	1.2E-05
Y-91m	3.3E-03	3.7E-02	1.1E-01	1.5E+00	8.6E-06
Y-91	5.7E-04	6.5E-03	1.9E-02	2.5E-01	2.7E-05
Y-93	3.8E-04	4.3E-03	1.3E-02	1.7E-01	8.3E-07
Zr-95	6.5E-04	7.3E-03	2.2E-02	3.0E-01	1.4E-05

TABLE 11.2-8

LIQUID WASTE PROCESSING SYSTEM COMPONENT DESIGN INVENTORIES - ORIGINAL LICENSING BASIs[a]

(Sheet 2 of 3)

Nuclide	Reactor Coolant Activity at 1% Fuel Defects (uCi/gm)	Tank Inventory (Ci)			
		Floor Drain Tank III	Waste Holdup Tank	Recycle Holdup Tank	Waste Monitor Tank
Nb-95	6.5E-04	7.4E-03	2.1E-02	3.0E-01	1.5E-05
Mo-99	7.5E-01	8.6E+00	2.5E+01	3.4E+02	9.7E-03
Tc-99m	6.9E-01	7.8E+00	2.2E+01	3.1E+02	1.1E-02
Ru-103	6.2E-04	7.1E-03	2.0E-02	2.8E-01	1.2E-05
Ru-106	2.6E-04	3.0E-03	8.4E-03	1.2E-01	5.8E-07
Ag-110m	1.4E-03	1.6E-02	4.5E-02	6.2E-01	3.1E-05
Te-129m	1.9E-02	2.2E-01	6.1E-01	8.5E+00	4.0E-04
Te-129	1.8E-02	2.0E-01	5.8E-01	8.3E+00	1.9E-06
Te-131m	2.6E-02	2.9E-01	8.3E-01	1.2E+01	2.1E-04
Te-131	1.2E-02	1.4E-01	3.9E-01	5.5E+00	3.4E-06
I-131	2.8E+00	3.2E+01	9.3E+01	1.2E+03	5.0E-03
Te-132	2.9E-01	3.2E+00	9.4E+00	1.3E+02	3.9E-03
I-132	2.8E+00	3.2E+01	9.1E+01	1.3E+03	3.1E-04

TABLE 11.2-8

LIQUID WASTE PROCESSING SYSTEM COMPONENT DESIGN INVENTORIES - ORIGINAL LICENSING BASIs[a]

(Sheet 3 of 3)

Nuclide	Reactor Coolant Activity at 1% Fuel Defects (uCi/gm)	Tank Inventory (Ci)			
		Floor Drain Tank III	Waste Holdup Tank	Recycle Holdup Tank	Waste Monitor Tank
I-133	4.2E+00	4.8E+01	1.4E+02	1.9E+03	2.4E-03
I-134	5.7E-01	6.5E+00	1.8E+01	2.5E+02	7.7E-08
Cs-134	6.8E+00	7.6E+01	2.2E+02	2.9E+03	1.6E-03
I-135	2.3E+00	2.7E+01	7.4E+01	1.1E+03	2.3E-04
Cs-136	2.9E+00	3.3E+01	9.3E+01	1.3E+03	6.0E-04
Cs-137	2.9E+00	3.4E+01	9.1E+01	1.3E+03	7.1E-04
Ba-140	4.2E-03	4.8E-02	1.4E-01	1.9E+00	8.1E-05
La-140	1.4E-03	1.6E-02	4.5E-02	6.2E-01	2.1E-05
Ce-141	6.3E-04	7.1E-03	2.0E-02	2.8E-01	1.3E-05
Ce-143	5.0E-04	5.7E-03	1.6E-02	2.3E-01	4.5E-06
Ce-144	4.4E-04	5.0E-03	1.5E-02	2.0E-01	9.7E-06
H-3	7.1E+00 (max)	7.9E+01	2.2E+02	3.2E+03	1.4E+01

TABLE 11.2-8

LIQUID WASTE PROCESSING SYSTEM COMPONENT DESIGN INVENTORIES - ORIGINAL LICENSING BASIS[a]

(Sheet 4 of 4)

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

TABLE 11.2-9
EXPECTED RELEASE CONCENTRATION VERSUS LIMITING CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 3)

Isotope	Expected Release Concentration[b] (uCi/gm)	Limiting Concentration[c] (uCi/gm)	
		Soluble	Insoluble
H-3	2.0E-01	3E-03	3E-03
Na-24	9.7E-06	2E-04	3E-05
Cr-51	3.4E-06	2E-03	2E-03
Mn-54	1.9E-06	1E-04	1E-04
Fe-55	1.4E-06	8E-04	2E-03
Fe59	3.4E-07	6E-05	5E-05
Co-58	5.3E-06	1E-04	9E-05
Co-60	6.3E-07	5E-05	3E-05
Ni-63	1.6E-09	3E-05	7E-04
Zn-65	5.9E-07	1E-04	2E-04
Sr-89	1.6E-07	3E-06	3E-05
Sr-90	2.3E-08	3E-07	4E-05
Sr-91	9.7E-08	7E-05	5E-05
Y-90	6.5E-09	2E-05	2E-05
Y-91	1.3E-08	3E-05	3E-05
Y-91m	6.6E-08	3E-03	3E-03
Y-93	4.1E-07	3E-05	3E-05
Zr-95	4.5E-07	6E-05	6E-05
Nb-95	3.3E-07	1E-04	1E-04
Nb-95m	3.2E-09	No Data	No Data

TABLE 11.2-9
EXPECTED RELEASE CONCENTRATION VERSUS LIMITING CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 3)

Isotope	Expected Release Concentration[b] (uCi/gm)	Limiting Concentration[c] (uCi/gm)	
		Soluble	Insoluble
Mo-99	4.4E-06	2E-04	4E-05
Tc-99m	4.1E-06	6E-03	3E-03
Ru-103	8.8E-06	8E-05	8E-05
Ru-106	2.1E-05	1E-05	1E-05
Rh-103m	8.1E-06	1E-02	1E-02
Ag-110m	1.5E-06	3E-05	3E-05
Te-129	1.4E-07	8E-04	8E-04
Te-129m	2.1E-07	3E-05	2E-05
Te-131	1.1E-07	No Data	No Data
Te-131m	6.3E-07	6E-05	4E-05
Te-132	1.2E-06	3E-05	2E-05
I-131	4.3E-06	3E-07	6E-05
I-132	1.2E-06	8E-06	2E-04
I-133	4.3E-06	1E-06	4E-05
I-134	2.4E-09	2E-05	6E-04
I-135	1.4E-06	4E-06	7E-05
Cs-134	2.6E-07	9E-06	4E-05
Cs-136	9.7E-09	9E-05	6E-05
Cs-137	2.3E-07	2E-05	4E-05
Ba-137m	1.1E-07	No Data	No Data

TABLE 11.2-9
EXPECTED RELEASE CONCENTRATION VERSUS LIMITING CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 3)

Isotope	Expected Release Concentration[b] (uCi/gm)	Limiting Concentration[c] (uCi/gm)	
		Soluble	Insoluble
Ba-140	1.3E-05	3E-05	2E-05
La-140	2.0E-05	2E-05	2E-05
Ce-141	1.6E-07	9E-05	9E-05
Ce-143	1.3E-06	4E-05	4E-05
Ce-144	5.1E-06	1E-05	1E-05
Pr-143	1.8E-07	5E-05	5E-05
Pr-144	4.5E-06	No Data	No Data
W-187	8.9E-07	7E-05	6E-05
Np-239	1.4E-06	1E-04	1E-04

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

b) Derived from Table 11A-1, assuming a discharge flow rate of 17,820 gal/day and no dilution due to Circulating Water.

c) From 10 CFR Part 20, Appendix B, Table II, Column 2. Design assessment made with the provision of 10CFR20.1-20.601.

TABLE 11.2-10
DESIGN RELEASE WITH 1-PERCENT FAILED FUEL - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 2)

Isotope	Reactor Coolant Activity at 1% Fuel Defects (uCi/gm)	Design Release Concentration[b] (uCi/gm)	Limiting Concentration[c] (uCi/gm)	
			Soluble	Insoluble
H-3	7.1E+0 (max)	7.1E-1	3E-03	3E-03
Cr-51	5.5E-3	6.0E-6	2E-03	2E-03
Mn-54	4.4E-4	5.2E-7	1E-04	1E-04
Fe-55	2.0E-3	2.3E-6	8E-04	2E-03
Fe-59	5.2E-4	5.9E-7	6E-05	5E-05
Co-58	1.5E-2	1.7E-5	1E-04	9E-05
Co-60	1.9E-3	2.3E-6	5E-05	3E-05
Sr-89	4.3E-3	4.9E-6	3E-06	3E-05
Sr-90	2.0E-4	2.3E-7	3E-07	4E-05
Sr-91	6.2E-3	6.3E-7	7E-05	5E-05
Y-91	5.7E-4	1.4E-6	3E-05	3E-05
Y-91m	3.3E-3	4.7E-7	3E-03	3E-03
Y-93	3.8E-4	4.4E-8	3E-05	3E-05
Zr-95	6.5E-4	7.5E-7	6E-05	6E-05
Nb-95	6.5E-4	7.7E-7	1E-04	1E-04
Mo-99	7.5E-1	5.2E-4	2E-04	4E-05
Tc-99m	6.9E-1	6.0E-4	6E-03	3E-03
Ru-103	6.2E-4	6.8E-7	8E-05	8E-05
Ru-106	2.6E-4	3.2E-8	1E-05	1E-05
Ag-110m	1.4E-3	1.6E-6	3E-05	3E-05
Te-129	1.8E-2	1.1E-7	8E-04	8E-04

TABLE 11.2-10
DESIGN RELEASE WITH 1-PERCENT FAILED FUEL - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 2)

Isotope	Reactor Coolant Activity at 1% Fuel Defects (uCi/gm)	Design Release Concentration[b] (uCi/gm)	Limiting Concentration[c] (uCi/gm)	
			Soluble	Insoluble
Te-129m	1.9E-2	2.1E-5	3E-05	2E-05
Te-131	1.2E-2	1.7E-7	No Data	No Data
Te-131m	2.6E-2	1.1E-5	6E-05	4E-05
Te-132	2.9E-1	2.0E-4	3E-05	2E-05
I-131	2.8E+0	2.7E-4	3E-07	6E-05
I-132	2.8E+0	1.6E-5	8E-06	2E-04
I-133	4.2E+0	1.3E-4	1E-06	4E-05
I-134	5.7E-1	4.0E-9	2E-05	6E-04
I-135	2.3E+0	1.2E-5	4E-06	7E-05
Cs-134	6.8E+0	8.5E-5	9E-06	4E-05
Cs-136	2.9E+0	3.2E-5	9E-05	6E-05
Cs-137	2.9E+0	3.6E-5	2E-05	4E-05
Ba-140	4.2E-3	4.2E-6	3E-05	2E-05
La-140	1.4E-3	1.1E-6	2E-05	2E-05
Ce-141	6.3E-4	6.7E-7	9E-05	9E-05
Ce-143	5.0E-4	2.3E-7	4E-05	4E-05
Ce-144	4.4E-4	5.1E-7	1E-05	1E-05

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

b) Derived from expected concentrations in Table 11.2-9. No credit is taken for dilution due to circulating water.

TABLE 11.2-1
DESIGN RELEASE WITH 1-PERCENT FAILED FUEL - ORIGINAL LICENSING BASIS[a]

(SHEET 3 OF 3)

c) From 10 CFR Part 20, Appendix B, Table II, Column 2. Design assessment made with the provisions of 10 CFR20.1-20.601.

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

This section describes the capabilities of the plant to control, collect, process, store and dispose of gaseous radioactive wastes generated as a result of normal operation including anticipated operational occurrences. The section discusses the design and operating features of the Gaseous Waste Processing System (GWPS) and the performance of other gas treatment and ventilation systems. Total gaseous releases from the plant for normal operation and the resulting offsite doses are also presented.

The radioactivity values presented in this section are the design basis values used for the design of the gaseous waste management system. As such they are considered historical and are not subject to future updating. The information is retained to avoid loss of original design basis. Actual radioactivity release quantities can be found in the annual radioactive effluent release reports submitted to the NRC.

11.3.1 DESIGN BASES

The gaseous waste systems are designed to collect, process, store and release gaseous wastes generated due to plant operations including anticipated operational occurrences. The systems are designed to assure that the release of gaseous effluents from the plant and expected offsite doses are as low as reasonably achievable as defined in the design objectives in Appendix I of 10CFR50. An evaluation of plant conformance to Appendix I is given in Appendix 11A. The gaseous systems have sufficient capacity and redundancy to meet discharge concentration limits of 10CFR20 during periods of design basis fuel leakage, as discussed in Section 11.3.3.

In addition, the GWPS meets the requirements of General Design Criterion 60 by providing long term holdup capacity, thus precluding the release of radioactive effluents during unfavorable environmental conditions. All gaseous effluent discharge paths are monitored for radioactivity, in compliance with General Design Criterion 64. Monitoring of radioactive effluents is discussed in Section 11.5.

The design of the GWPS is based on continuous operation of the Nuclear Steam Supply System assuming that fission products associated with 1 percent of the core power generation are available for leakage from the fuel into the coolant. This condition is assumed to exist over the life of the plant.

A discussion of system design, component design and materials of construction is contained in Sections 11.3.2.1, 11.3.2.1.1, and 11.3.2.1.2. System operation during normal operation and anticipated operational occurrences is described in Section 11.3.2.1.3.

The expected and design inventories of nuclides in the GWPS components are provided in Section 11.3.2.1.4. Design provisions incorporated to control the release of radioactive materials in gaseous effluents are discussed in Sections 11.3.2.1.1 and 11.3.2.1.2. The GWPS is designed to maintain gas composition outside the range of flammable and explosive mixtures as described in Section 11.3.2.1.1 and 11.3.2.1.2.

11.3.2 SYSTEMS DESCRIPTION

This section describes the design and operating features of the GWPS. The performance of the GWPS and other plant gaseous waste management systems with respect to the collection and

control of radioactive gases is also discussed in this section. Detailed descriptions of plant ventilation systems and the condenser vacuum system are presented in Sections 9.4 and 10.4, respectively.

11.3.2.1 Gaseous Waste Processing System Design

The piping and instrumentation diagrams for the GWPS are shown in the Flow Diagram Volume as indexed against Figure 11.3-1. These diagrams indicate safety classes for all components and piping.

The GWPS is shared between Comanche Peak Nuclear Power Plant (CPNPP) Unit 1 and Unit 2. The main flow path in the GWPS is a closed loop comprised of two waste gas compressors, two catalytic hydrogen recombiners, eight gas decay tanks for normal power service and two gas decay tanks for service at shutdown and startup. The eight gas decay tanks used for normal power service can also be used to function as shutdown gas decay tanks at shutdown and startup. The system also includes a gas decay tank drain pump, four gas traps, and a waste gas drain filter. All of the equipment is located in the primary Auxiliary Building.

The GWPS stores fission gases removed from the Reactor Coolant System (RCS). This reduces the escape of fission gases from the RCS during maintenance operations or through equipment leakage.

The primary location from which radioactive gases are removed from the RCS is the volume control tank. Smaller quantities are received via the vent connections, from the reactor coolant drain tank, the pressurizer relief tank, and the recycle holdup tanks. The waste and recycle evaporator gas strippers are normally vented to the Auxiliary Building exhaust.

The largest contributor to the nonradioactive gas accumulation is helium generated by a boron-10 (n,α) lithium-7 reaction in the reactor core. The second largest contributors are impurities in the bulk hydrogen and oxygen supplies. Stable and long-lived isotopes of fission gases also contribute small quantities to the system gas accumulation.

Gases from the above sources should be contained as long as practical, thus the discharges from the GWPS to the environment for normal plant operation should occur infrequently.

Operation of the system is such that fission gases are distributed throughout the eight normal operation gas decay tanks. Separation of the GWPS gaseous inventory in several tanks reduces the amount of fission gases that would be released in the event of a gas decay tank rupture. Radiological consequences of such a postulated rupture are discussed in Chapter 15.

The GWPS also provides capacity for holdup of gases generated during reactor shutdown. A portion of the gas from shutdowns is typically contained in one of the shutdown gas decay tanks. The second shutdown tank is normally at low pressure and is used to accept relief valve discharges from the normal operation gas decay tanks.

11.3.2.1.1 Component Design

Gaseous waste processing equipment design parameters are given in Table 11.3-1. This system performs no function related to the safe shutdown of the plant. Component safety classes,

American Society of Mechanical Engineers (ASME) Code and seismic design are discussed in Chapter 3. Except for flanged joints, all-welded construction is used whenever it is practical.

Pressure retaining components of gaseous waste processing equipment meet the design guidance, including quality assurance requirement, as outlined in Branch Technical Position ETSB 11-1 and FSAR Table 17A-1.

The waste gas compressor is a water sealed centrifugal displacement unit which maintains continuous circulation of nitrogen around the waste gas loop. The compressor is provided with a mechanical shaft seal to minimize water leakage.

The catalytic recombiner recombines hydrogen brought into the GWPS with oxygen to form water. This is accomplished by adding a controlled amount of oxygen to the recombiner which reacts with the hydrogen as the gas flows through a catalyst bed. The catalyst is 0.5% paladium on a Kaolin Base. The recombiner is designed with a low flow rate across the catalyst bed to maintain the reaction within the recombiner. The control system for the recombiner maintains an oxygen lean mixture to preclude the possibility of a hydrogen explosion. This is further discussed in Section 11.3.2.1.2.

The valves in this system are designed to minimize outleakage of radioactive gases.

Relief valves have soft seats and the normal operating pressures are typically two-thirds of the relief valve set pressure. The relief valves of the major components discharge to the shutdown tanks. This permits decay and controlled disposal of all expected discharges. The relief valves are designed to relieve full flow from both waste gas compressors.

The gas decay tank drain pump directs water from the gas decay tanks (due to condensation or maintenance) to the waste holdup tank or Recycle Holdup Tanks. It is used when there is insufficient pressure in the gas system to drive the fluid. All parts of the pump in contact with the drain water are of austenitic stainless steel. The pump is a canned motor type.

The waste gas drain filter is a disposable cartridge filter provided to prevent particulate matter from migrating into other systems. All parts of the filter in contact with the drain water are of austenitic stainless steel.

The four waste gas traps are designed to prevent undissolved gases from leaving the GWPS when water is drained off the system.

11.3.2.1.2 Instrumentation and Control Design

The main system instrumentation is described in Table 11.3-2 and shown on the piping and instrumentation diagrams in the Flow Diagram Volume.

The instrumentation readout is located mainly on the Waste Processing System (WPS) panel in the Auxiliary Building. Some instruments are read where the equipment is located. All alarms are shown separately on the WPS panel and further relayed to one common WPS annunciator on the main control board of the plant. Where suitable, instrument lines are provided with diaphragm seals to prevent fission gas outleakage through the instrument.

The compressors are interlocked with the seal water inventory in the moisture separators and trips off on either high or low moisture separator level. During normal operation the proper seal water inventory is maintained automatically.

The catalytic recombiner system is designed for automatic operation. Each package includes two online gas analyzers, one to measure hydrogen in and oxygen in, and one to measure hydrogen out and oxygen out, which are the primary means of recombiner control.

Process gas flow rate is measured upstream of the recombiner preheater. Local pressure gauges indicate pressure at the recombiner inlet and the oxygen supply pressure.

The following is a summary of the controls and alarms incorporated to maintain the gas composition outside the range of flammable and explosive mixtures. Specific system operational requirements/limitations are specified in the CPNPP Technical Requirements Manual.

1. If the hydrogen concentration in the recombiner feed reaches 9 percent by volume, a hi hydrogen alarm sounds, the oxygen feed is terminated, and the volume control tank hydrogen purge flow is terminated.

2. If the oxygen concentration in the recombiner feed reaches the maximum capacity of the catalytic hydrogen recombiner an alarm sounds and oxygen feed flow is limited so that no further increase in flow is possible. This control maintains the normal system oxygen concentration at a level which is below the flammable limit for hydrogen-oxygen mixtures. Oxygen concentration is always limited to $\leq$4% in this system.

3. If hydrogen in the recombiner discharge is excessive an alarm sounds. This alarm warns of high hydrogen feed, possible reactor malfunction, or loss of oxygen feed.

4. If oxygen in the recombiner discharge becomes excessive an alarm sounds and oxygen feed is terminated. This control maintains system oxygen lean in case of reactor malfunction.

5. On low flow through the recombiner, oxygen feed is terminated.

6. High discharge temperature from the cooler-condenser (downstream from the recombiner reactor) will terminate oxygen feed.

7. High temperature indication by any one of six thermocouples in the catalyst bed will terminate oxygen feed.

8. High temperature indication at the recombiner reactor discharge will terminate oxygen feed to the recombiner.

11.3.2.1.3 System Operation

Normal Operation

During normal power operation, nitrogen gas and fission gases are typically circulated around the GWPS loop by one of the two compressors. Hydrogen gas is introduced to the volume control tank where it is mixed with fission gases stripped from the reactor coolant by the action of

the volume control tank letdown line nozzle spray. The gas stream may then be vented from the volume control tank into the circulating nitrogen stream in the waste gas system, at the compressor suction.

The resulting mixture of nitrogen, hydrogen and fission gases is pumped by one of the compressors to one of the two catalytic hydrogen recombiners where enough oxygen is added to react with and reduce the hydrogen to a low residual level. Water vapor formed in the recombiner by the hydrogen-oxygen reaction is condensed and removed, and the cooled gas stream (now composed primarily of nitrogen and fission gases) is discharged from the recombiner, routed through a gas decay tank, and sent back to the compressor suction to complete the loop circuit. Depending on gas decay tank pressure the waste gas may be pumped by the compressor to a gas decay tank prior to processing by the hydrogen recombiner.

Only one gas decay tank is valved into the waste gas loop at any time. By switching tanks at regular intervals, the radioactive gas inventory in the system is distributed through all the gas decay tanks.

Startup

At plant startup, the system is first flushed free of air and filled with nitrogen. One compressor, one recombiner and one Shutdown Gas Decay Tank are in service. The reactor is at the cold shutdown condition. In the twin unit CPNPP, the volume control tank purge from the second unit is terminated while the unit of interest is started up. Hydrogen is charged to the volume control tank and the volume control tank vent gas mixes with the circulating nitrogen stream in the GWPS. This circulating mixture enters the compressor suction, passes through the recombiner and Shutdown Gas Decay Tank and returns to the compressor suction. When the RCS hydrogen concentration is within operating specifications, the Shutdown Gas Decay Tank is isolated and the gas flow directed to one of the Gas Decay Tanks provided for normal power operation. Hydrogen purge flow from the second unit is then restarted. Gases accumulated in the Shutdown Gas Decay Tank will be retained for reuse during hydrogen stripping from the RCS.

Shutdown and Degassing of the Reactor Coolant

Plant shutdown operations are essentially startup operations in reverse sequence. The volume control tank hydrogen purge is maintained until after the reactor is shutdown and coolant fission gas concentrations have been reduced to specified level. During this operation hydrogen purge flow may be increased to speed up coolant degassing. Next the hydrogen purge to the unit still at power is stopped and the gas decay tank in service for normal power operation is valved out. A nitrogen purge from the Shutdown Gas Decay Tank to the volume control tank of the shutdown unit is begun. This tank is placed in the process loop at the compressor discharge so that the gas mixture from the volume control tank vents to the compressor suction, passes through the Shutdown Gas Decay Tank and to the recombiner where hydrogen is removed and returned to the compressor suction. The nitrogen purge continues until reactor coolant hydrogen concentration in the volume control tank reaches the required level. At this point the Shutdown Gas Decay Tank may be isolated and the GWPS returned to its previous operating mode for the unit still at power. Depending upon system pressure the shutdown and degassing flow path may be volume control tank to waste gas compressor to hydrogen recombiner to shutdown gas decay tank and returning to the volume control tank.

During the first plant cold shutdown, nitrogen was charged to the volume control tank to strip hydrogen from the reactor coolant. The resulting accumulation of nitrogen in the Shutdown Gas Decay Tank is accommodated by allowing the tank pressure to increase. During subsequent shutdowns, however, there should be minimal additional accumulation since the gas from the first shutdown will be reused.

11.3.2.1.4 Radioactivity in the Gaseous Waste Processing System

Design Basis Case

Table 11.3-3 shows the accumulated activity in the GWPS after 40 years of plant operation using the design basis assumptions given in Table 11.1-1 but assuming 100 percent stripping efficiency. The values in this table are thus the maximum accumulated fission product activity in the GWPS. Figure 11.3-4 shows gaseous fission product accumulation as a function of time over the life of the plant, using design basis assumptions. As is seen in the figure, the increase in activity over the plant life is due to the build up of krypton-85. Other gaseous isotopes reach equilibrium in approximately 30 days.

Realistic Case

Plant activity release for environmental impact evaluations are calculated using realistic assumptions. Table 11.3-4 is the expected 40 year isotopic inventory in the GWPS, using the realistic basis assumptions of Table 11.1-3.

These values are used in calculating the radioactive release from the GWPS (refer to Section 11.3.3). The process flow diagram for the GWPS is shown in the Flow Diagram Volume as indexed against Figure 11.3-1. Table 11.3-5 presents the gas composition, flow rate and isotopic concentration at each point indicated in the process flow diagram. Isotopic inventories of major components and isotopic concentrations of liquid streams in the GWPS are also shown in the table.

For purposes of this table and release calculations, it was assumed that the gas decay tank in the GWPS circulating loop is switched on a daily basis.

11.3.2.2 Ventilation System Design

Details of the plant ventilation system, including the atmospheric cleanup filtration system are presented in Section 9.4.

11.3.2.3 Condenser Vacuum System Design

Details of the Condenser Vacuum System are presented in Section l0.4.2.

11.3.3 ESTIMATED RELEASES

This section describes the estimated gaseous release from the plant for normal operation and anticipated operational occurrences. In general, the following major sources will contribute to the expected releases of radioactive materials in gaseous effluents resulting from normal operations, including anticipated operational occurrences:

1. Leakage from the GWPS.

2. Reactor coolant leakage to the Containment.

3. Reactor coolant leakage to the Auxiliary Building.

4. Secondary side steam and water leakage to the Turbine Building.

5. Discharge from the condenser air ejector.

6. Venting during equipment drainage and sampling.

7. Venting of WGDT for maintenance.

11.3.3.1 Estimated Release from the Gaseous Waste Processing System

The GWPS is designed to provide a holdup of waste gases such that frequent, scheduled releases are not anticipated. When operationally planned releases of radioactive gases from the GWPS become necessary, such releases will be permitted only under controlled conditions subject to the limitations of 10CFR20 and 10CFR50, Appendix I. As shown in Figure 11.3-5, Luminant Power has agreed to the stipulation that CPNPP will be operated such that dose commitments from planned gaseous radioactive releases are as low as is reasonably achievable (ALARA). The estimated annual release from the GWPS is calculated based on an assumed leakage from the system. Leakage rates have been estimated for all components in the system and are shown in Table 11.3-6. Based on this leak rate and the expected system activity inventory shown in Table 11.3-4 the annual gaseous release from the system is presented in Table 11.3-7.

11.3.3.2 Estimated Releases from Ventilation Systems

The estimated annual releases from the ventilation system are presented in Appendix 11A, Table 11A.4.

TABLE 11.3-1
GASEOUS WASTE PROCESSING SYSTEM COMPONENT DATA

Waste Gas Compressors	
Type	Centrifugal
Quantity	2
Design pressure (psig)	150
Design temperature (°F)	180
Operating temperature (°F)	70 to 130
Design suction pressure, N_2 at 130°F (psig)	0.5
Design discharge pressure (psig)	110
Design flow, N_2 at 130°F (scfm)	40
Gas Decay Tanks	
Type	Vertical
Quantity	10
Design pressure (psig)	150
Design temperature (°F)	180
Volume, each (ft^3)	600
Material of construction	Carbon Steel
Recombiners	
Type	Catalytic
Quantity	2
Design inlet pressure (psig)	110
Design inlet temperature (°F)	140
Design flow rate (scfm)	50
Design hydrogen recombiner rate (scfm)	3.0
Design discharge pressure (psig)	25
Design discharge temperature (°F)	140
Material of construction	Stainless Steel

TABLE 11.3-2
GASEOUS WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[(a)]

(Sheet 1 of 3)

Channel Number	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location Of Readout
Flow Instrumentation							
FIA-1094	Volume control tank discharge flow	150	140	0.0-1.2 scfm	1.2 scfm	-	WPS panel
QIA-1091	Gas decay tank water flush	150		0-30 gpm	30 gpm (adjustable)	-	Local
HIC-1094	Volume control tank purge control	150		0-100 pct	None	Manual control (normal flow 0.7 scfm)	WPS panel
Pressure Instrumentation							
PI-1031	Moisture separator	150	180	0-100 psig		-	Local
PI-1033	Moisture separator	150	180	0-100 psig		-	Local
PIA-1036	Gas decay tank no. 1	150	180	0-150 psig 0-30 psig	100 psig 20 psig	-	WPS panel
PIA-1037	Gas decay tank no. 2	150	180	0-150 psig 0-30 psig	100 psig 20 psig	-	WPS panel
PIA-1038	Gas decay tank no. 3	150	180	0-150 psig 0-30 psig	100 psig 20 psig	-	WPS panel

TABLE 11.3-2
GASEOUS WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[a]

(Sheet 2 of 3)

Channel Number	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location Of Readout
PIA-1039	Gas decay tank no. 4	150	180	0-150 psig 0-30 psig	100 psig 20 psig	-	WPS panel
PIA-1052	Gas decay tank no. 5	150	180	0-150 psig 0-30 psig	100 psig 20 psig	-	WPS panel
PIA-1053	Gas decay tank no. 6	150	180	0-150 psig 0-30 psig	100 psig 20 psig	-	WPS panel
PIA-1054	Gas decay tank no. 7	150	180	0-150 psig 0-30 psig	100 psig 20 psig	-	WPS panel
PIA-1055	Gas decay tank no. 8	150	180	0-150 psig 0-30 psig	100 psig 20 psig	-	WPS panel
PIA-1056	Gas decay tank no. 9	150	180	0-150 psig 0-30 psig	80 psig	-	WPS panel
PIA-1057	Gas decay tank no. 10	150	180	0-150 psig 0-30 psig	80 psig 20 psig	-	WPS panel
PIA-1065	Hydrogen supply header	150	180	0-150 psig	90 psig	-	WPS panel
PIA-1066	Nitrogen supply header	150	180	0-150 psig	53 psig	-	WPS panel
PICA-1092	Compressor suction header	150	180	2 psi vac. 2 psig	0.5 psi vac	0.5 psi vac.	WPS panel
PI-1093	Gas decay tank makeup water	240	180	0-160 psig	-	Local	

TABLE 11.3-2
GASEOUS WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS[a]

(Sheet 3 of 3)

Channel Number	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location Of Readout
PI-1094	Volume control tank discharge pressure	150	250	0-20 psig	-	-	Local
Level Instrumentation							
LICA-1030	Compressor moisture separator	150	180	0-100% (6-33 in wc)	Low-7 inch High - 21.1 inch	21.1" to 16" in wc 16" to 14" in wc 14" to 11" in wc 11" to 7" in wc	WPS panel and local
LICA-1032	Compressor moisture separator	150	180	0-100% (6-33 in wc)	Low-7 inch High - 21.1 inch	21.1 to 16" in wc 16" to 14" in wc 14" to 11" in wc 11" to 7" in wc	WPS panel and local

a)
F - Flow
Q - Water Integrator
P - Pressure
T - Temperature
L - Level
R - Radiation
I - Indication
C - Control
A - Alarm

TABLE 11.3-3
DESIGN BASIS ACCUMULATED RADIOACTIVITY IN THE GASEOUS WASTE PROCESSING SYSTEM AFTER FORTY YEARS OPERATION (BASED ON OPERATION OF TWO UNITS) - ORIGINAL LICENSING BASIS [a]

Isotope	Activity (Curies) At Plant Shutdown
Kr-85	78,000
All other noble gases	
Kr-85m	116
Kr-87	10.8
Kr-88	110
Xe-131m	1240
Xe-133	132,000
Xe-133m	6600
Xe-135	960
Xe-135m	3.8
Xe-138	0.26

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

This table is based on 40 years continuous operation with 1 percent fuel defect. Power assumed to be 3565 MWt. The data are based on the design basis assumptions listed in Table 11.1-1 but assuming 100 percent stripping efficiency.

TABLE 11.3-4
REALISTIC ACCUMULATED RADIOACTIVITY IN THE GASEOUS WASTE PROCESSING SYSTEM AFTER FORTY YEARS OPERATION (BASED ON OPERATION OF TWO UNITS) - ORIGINAL LICENSING BASIS[(a)]

Isotope	Activity (Curies) At Plant Shutdown
Kr-85	15,540
All other noble gases	
Kr-85m	2.86
Kr-87	0.208
Kr-88	2.58
Xe-131m	63.6
Xe-133	7,480
Xe-133m	60.0
Xe-135	21.6
Xe-135m	0.0022
Xe-138	0.00646
Iodine	
I-131	0.1864
I-132	0.0008
I-133	0.0284
I-134	0.000126
I-135	0.0046

Inventories are based on reactor coolant concentrations given in Table 11.1-4. The table is based on 40 years continuous operation. Power assumed to be 3565 MWt. The data are based on a volume control tank purge rate of 0.7 scfm, a 40 percent stripping efficiency and the stripping fractions listed in Table 11.1-3.

(a) Historical, not subject to future updating. Has been retained to preserve original design basis.

TABLE 11.3-5
PROCESS PARAMETERS FOR GASEOUS WASTE PROCESSING SYSTEM[a] - ORIGINAL LICENSING BASIS[b]

(Sheet 1 of 4)

Description	Flow (scfm)	N_2 (%)	H_2 (%)	Isotopic Concentration[c] (µc/cc)						
				Kr-85[d]	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-133m	Xe-135
Volume control tank purge	1.4	0	100	$2.67x10^{-2}$	$1.78x10^{-1}$	$4.52x10^{-2}$	$2.56x10^{-1}$	$1.63x10^{1}$	$3.12x10^{-1}$	$6.50x10^{-1}$
Gas decay tank discharge to compressor	40	99.9	0.1	$1.70x10^{1}$	$1.51x10^{-1}$	$1.10x10^{-2}$	$1.37x10^{-1}$	$7/48x10^{1}$	3.17	1.14
Compressor suction	41.4	96.6	3.4	$1.64x10^{1}$	$1.52x10^{-1}$	$1.22x10^{-2}$	$1.41x10^{-1}$	$7.28x10^{1}$	3.08	1.13
Compressor discharge to Recombiner	41.4	96.6	3.4	$1.64x10^{1}$	$1.52x10^{-1}$	$1.22x10^{-2}$	$1.41x10^{-1}$	$7.28x10^{1}$	3.08	1.13
Recombiner discharge to gas decay tanks	40	99.9	0.1	$1.70x10^{1}$	$1.58x10^{-1}$	$1.26x10^{-2}$	$1.45x10^{-1}$	$7.54x10^{1}$	3.19	1.17
Miscellaneous vents, evaporators, RCDT, recycle holdup tank educator	Neg.	0	100	0	0	0	0	0	0	0
Recombiner oxygen supply	0.70	0	0	0	0	0	0	0	0	0
Recombiner calibrating gas	0.004	94	6	0	0	0	0	0	0	0
Recombiner calibrating gas	0.004	94	6	0	0	0	0	0	0	0

TABLE 11.3-5
PROCESS PARAMETERS FOR GASEOUS WASTE PROCESSING SYSTEM[a] - ORIGINAL LICENSING BASIS[b]

(Sheet 2 of 4)

Waste gas system nitrogen supply	0	100	0	0	0	0	0	0	0	0

Description	Flow (scfm)	N_2 (%)	H_2 (%)	Isotopic Concentration[c] (µc/cc)						
				Kr-85[d]	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-133m	Xe-135
NSSS nitrogen supply	0	100	0	0	0	0	0	0	0	0
Nitrogen relief to plant vent	0	100	0	0	0	0	0	0	0	0
NSSS hydrogen supply	1.4	0	100	0	0	0	0	0	0	0
Volume control tank hydrogen	1.4	0	100	0	0	0	0	0	0	0
Hydrogen relief to plant vent	0	0	100	0	0	0	0	0	0	0
Waste gas discharge to plant vent	0	100	0	1.70×10^{1}	0	0	0	0	0	0
Recycle gas to volume control tank	0	100	0	0	0	0	0	0	0	0
Pressurizer relief tank vent and return	0	100	0	0	0	0	0	0	0	0
Shutdown tank relief	0	100	0	0	0	0	0	0	0	0

TABLE 11.3-5
PROCESS PARAMETERS FOR GASEOUS WASTE PROCESSING SYSTEM[a] - ORIGINAL LICENSING BASIS[b]

(Sheet 3 of 4)

Description	Flow (scfm)	N_2 (%)	H_2 (%)	Isotopic Concentration[e] (µc/cc) Kr-85[d]	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-133m	Xe-135
Liquid Streams										
Waste gas compressor drain		4.37	4.05×10^{-2}	3.24×10^{-3}	3.74×10^{-2}	1.60×10^{1}	6.77×10^{-1}	2.48×10^{-1}		
Recombiner drain		3.38	3.14×10^{-2}	2.51×10^{-3}	2.90×10^{-2}	1.24×10^{1}	5.25×10^{-1}	1.92×10^{-1}		
Gas decay tank drains		1.19	1.06×10^{-2}	7.72×10^{-4}	9.58×10^{-3}	4.34	1.84×10^{-1}	6.63×10^{-2}		
System drains to volume control tank		1.74	1.58×10^{-2}	1.21×10^{-3}	1.44×10^{-2}	6.36	2.69×10^{-1}	9.77×10^{-2}		
Recombiner reactor makeup water		0	0	0	0	0	0	0		
Compressor makeup water		0	0	0	0	0	0	0		

Description	Flow (scfm)	N_2 (%)	H_2 (%)	Component Inventory (Curies) Kr-85[d]	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-133m	Xe-135
Compressor	4	96.6	3.4	7.80	6.96×10^{-2}	5.06×10^{-3}	6.28×10^{-2}	3.44×10^{1}	1.46	5.26×10^{-1}
Recombiner	4	99.9	0.1	5.84	5.21×10^{-2}	3.79×10^{-3}	4.70×10^{-2}	2.57×10^{1}	1.09	3.93×10^{-1}

TABLE 11.3-5
PROCESS PARAMETERS FOR GASEOUS WASTE PROCESSING SYSTEM[a] - ORIGINAL LICENSING BASIS[b]

(Sheet 4 of 4)

Description	Flow (scfm)	N_2 (%)	H_2 (%)	Component Inventory (Curies)						
				Kr-85[d]	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-133m	Xe-135
Gas decay tank	600	99.9	0.1	1.93×10^{3}	2.74	1.99×10^{-1}	2.47	1.35×10^{3}	5.74×10^{1}	2.07×10^{1}
Total system				1.55×10^{4}	2.86	2.08×10^{-1}	2.58	7.48×10^{3}	6.00×10^{1}	2.16×10^{1}

Basis

Power level - 3565 Mwt per unit

Gas decay tanks[f] - 8

Operating interval - 1 day

Stripping efficiency - 0.4

a) Based on GWPS isotopic inventories from Table 11.3-4 and reactor coolant activities from Table 11.1-4.

b) Historical, not subject to future updating. Has been retained to preserve original design basis.

c) Concentrations in µc/cc of gas at atmospheric pressure and 140°F.

d) Kr-85 concentrations are maximum values, but do not occur simultaneously with other isotope maximum concentrations.

e) Concentrations in µc/cc liquid at room temperature.

f) Includes two shutdown tanks.

TABLE 11.3-6
ESTIMATED LEAKAGE RATES FROM MAJOR GASEOUS WASTE
PROCESSING SYSTEM COMPONENTS

	scf/year
Valves	~45
Compressors	~10
Tanks and Valves	~10
Pipe Flanges	~20
Total	~85[(a)]

a) The above value of 85 scf/year has been rounded off to 100 scf/year which has been used in the calculation of expected releases from the GWPS

TABLE 11.3-7
ESTIMATED GASEOUS WASTE PROCESSING SYSTEM RELEASES DUE TO LEAKAGE - ORIGINAL LICENSING BASIS[a]

Isotopes	Leakage (Curies/year)
Kr-85	92
Kr-85m	0.44
Kr-87	0.032
Kr-88	0.40
Xe-133	218
Xe-133m	2.6
Xe-135	3.32
Xe-135m	0
Xe-138	0
I-131	4.8×10^{-3}
I-133	2.4×10^{-3}
I-135	7.2×10^{-4}

This assumes operation of two units with reactor coolant inventories given in Table 11.1-4 and a leakage rate of 100 standard cubic feet per year from the GWPS. The stripping efficiency used for noble gases in the volume control tank is 40 percent. Separation factor for iodines in the volume control tank is 100.

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

11.4 SOLID WASTE MANAGEMENT SYSTEM

The solid waste management system (SWMS) is designed to control, collect, condition, handle, process, package, and temporarily store, prior to offsite shipment, solid radioactive waste generated as a result of normal operation, including anticipated operational occurrences. The SWMS consists of ATCOR system components (see Section 11.4.2.1) and the waste baling system (WBS). The WBS has a baler located in the Fuel Building. An additional waste baler was abandoned in place in the Unit 1 Containment Building.

11.4.1 DESIGN BASES

11.4.1.1 Design Objectives

The design objectives of the waste baler meet the requirements of 10 CFR Parts 20, 50, 61 and 71 and United States Department of Transportation (DOT) Hazardous Materials Regulation 49 CFR Parts 170 through 178.

Connections have been provided to allow for the bulk disposal of wastes to a truck mounted or mobile waste processing system. These connections supply waste from the Chemical Drain Tank, Waste Conditioning Tank, the NSSS Spent Resin Transfer System, and the Steam Generator Blowdown Spent Resin Transfer System.

The WBS consists of a compactor type-baler which may be used to package low-radiation-level compressible wastes such as paper, disposable clothing, rags, towels, floor coverings, shoe covers, plastics, cloth smears, and respirator filters in 55-gal drums. These wastes are products of plant operation and maintenance.

11.4.1.2 Design Criteria

The input to the SWMS is from various sources. The incompressible solids are packaged in suitable containers while the compressible solid wastes are collected and compressed (baled) into containers suitable for disposal or sent offsite for vendor processing. The various sources, quantities, and activity levels are detailed in Table 11.4-1. Maximum volumes may be produced as a result of excessive equipment leakage, steam generator tube leakage, and so forth.

The principal nuclides shipped from the plant site include the following:

Iodine-131	Iron-59
Cesium-134	Manganese-54
Cesium-136	Manganese-56
Cesium-137	Molybdenum-99
Cobalt-58	Strontium-89
Cobalt-60	Strontium-90
Iron-55	Chromium-51
Hydrogen-3	

The seismic design classification of the Fuel Building housing the SWMS is discussed in Section 3.2. The seismic design and equipment design codes for the SWMS components and piping are in accordance with ETSB Technical position 11-1. See Appendix 17A.

11.4.2 SYSTEM DESCRIPTION

11.4.2.1 Solid Waste System Description

The ATCOR Radwaste Solidification System originally installed at CPNPP is not used and many of the components have been removed from the facility. Only those components necessary for the collection and conditioning of the various waste streams prior to discharge via the bulk disposal connections on elevation 810' of the Fuel Building are utilized.

The major ATCOR system components which are utilized include the Waste Conditioning Tank, Waste Feeder Pumps, Emergency Waste Return Pump, Powdex Transfer Pump, Chemical Addition Tank and their associated valves, piping and controls.

11.4.2.2 Spent Filter Cartridge Assembly Processing

Transfer of expended filter cartridge assemblies from the filter housing to the waste processing area may be accomplished by means of a filter transfer cask. The base of the cask is removed and the cask positioned above the filter housing. The cask hoist grapple is lowered to engage the filter cartridge assembly and the cartridge raised into the shielded cavity. The base of the transfer cask is replaced, then the cask is moved by monorail to the filter drop zone hatchway where it is lowered into the Fuel Building. Other methods for transferring filters may be used predicated on the filter dose rate and ALARA considerations. For example, the filter transfer cask hoist may be used to remove the filter and lower it into the fuel building and into a smaller shielded container, without the use of the filter transfer cask; or the filter may be manually removed and transported to the fuel building.

Filters may be processed on site to comply with state, federal and burial site transportation and disposal regulations or they may be shipped to a vendor for offsite processing.

11.4.2.3 Large Solid Waste Materials and Equipment

Large waste materials and special equipment that have been neutron activated during reactor operation (e.g., core components) are handled and packaged in a safe manner on a case-by-case basis.

11.4.2.4 Waste Baling System Description

Processing of compressible low-radiation-level solid wastes for disposal may be accomplished onsite using the fuel building baler to compact the waste into 55 gallon drums, or waste may be packaged in suitable transportation containers and shipped to an offsite vendor for immediate processing and volume reduction prior to disposal.

The Fuel Building baler’s shroud is ducted to the plant ventilation system to remove dust or particles that may be emitted from the drum during compression of the wastes.

This provision eliminates any potential hazard from airborne radioactivity. In addition, the assembly incorporates a fail-safe switch that does not permit baler operation with the baler door open.

11.4.2.5 Component Description

All components which are located in the Fuel Building are at elevations of 810, 822, and 840 feet.

Associated components of the SWMS are as follows:

1. Containers

   All waste that is processed at CPNPP for disposal shall be packaged in strong, tight containers meeting all applicable DOT, NRC and burial site requirements pertaining to the storage, shipment, and burial of radioactive waste.

   Any waste packaged at CPNPP for shipment to an intermediate processor shall be packaged in strong, tight containers meeting all applicable DOT and NRC requirements pertaining to storage and transportation of waste.

2. Storage Shields

   The storage shields may be used to protect personnel from radiation exposure during onsite storage and other operational handling of the filled containers.

3. Filter Transfer Cask

   One or more filter transfer casks are provided to be used as a shielding and carrier vehicle to protect personnel from radiation exposure while transferring spent filter cartridges from the filter housing to the waste processing area. Movement of a filter transfer cask is by a monorail. Each cask is provided with a removable drip pan to collect any dripping liquid from the filter cartridges. Each cask is designed with a stainless steel interior to facilitate washdown and decontamination.

4. Baler

   The baler is a commercially available assembly used in conjunction with standard 55-gal drums which receive the low radiation level, solid, compressible wastes.

11.4.2.6 Other Design Features

1. Process Control Program (PCP)

   The PCP contains or makes reference to the current formulas, sampling, analyses, tests, and determinations made to ensure that processing and packaging of wet solid radioactive waste based on demonstrated processing of actual or simulated wastes will be accomplished in such a way to ensure compliance with federal and state regulations, burial site criteria, and other requirements governing the disposal of radioactive waste.

Waste processing is performed by a mobile processing vendor. The Luminant Power Process Control Program requires that the vendor operate in accordance with a process control program and procedures which have been reviewed and approved by Luminant Power. Additionally, any vendor selected to provide waste processing services or products used to achieve the 10 CFR 61 stability requirements shall have documentation demonstrating compliance with 10CFR61 stability requirements.

The primary purposes of the Luminant Power Process Control Program are as follows:

a. The PCP defines the expected wet radioactive waste streams to be processed at CPNPP. These waste streams may include bead and powdered resins, cartridge filters, evaporator concentrates, sludges from tanks and sumps, and miscellaneous liquids.

b. The PCP establishes criteria for waste processing that must be addressed in topical reports, process control programs, and/or procedures of vendors selected for waste processing. Such criteria include establishment of bounds for critical parameters and operating limits such as chemical constituents, pH, concentration of radioactive materials, boric acid content, etc.

c. The PCP establishes requirements for prequalification testing of each waste stream.

d. The PCP establishes requirements for sampling of waste streams to provide data necessary for estimating curie content and classifying the waste, and to ensure that the waste stream parameters are within the bounds for critical parameters established in the vendor's PCP.

e. The PCP establishes the testing requirements for each processing method to ensure that the 10CFR61 waste form and stability requirements and the applicable NRC burial site criteria are achieved.

f. The PCP establishes the administrative controls and quality assurance activities for waste processing required to ensure compliance with applicable regulations and requirements.

2. Overflow of Tanks

The Waste Conditioning Tank is provided with an ultrasonic level sensing device which provides level indication over the range of tank operating levels. In addition, level detectors are provided which provide interlock inputs for waste conditioning tank high and low levels. The high level interlock alarms and prevents any restart of waste stream addition to the Waste Conditioning Tank while the low level interlock stops the tank agitator.

3. Tanks Using Compressed Gases

Compressed gas is not directly used in any SWMS tank. However, the Waste Conditioning Tank is vented to the plant ventilation system since bead resin slurry is transported to the waste conditioning tank from the spent resin storage tanks by means of

nitrogen gas pressure. The volume and flow rate of the gas used for transferring one batch is estimated to be 1200 scf at 50 scfm. The expected radionuclide concentration of the vent gases will be negligible. The treatment provided (atmospheric cleanup system) for the vent gases is described in Section 9.4.

11.4.2.7 Packaging, Storage, and Shipment

1. Packaging

The SWMS product is a container of radioactive material packaged in accordance with all applicable NRC, DOT and disposal site requirements.

2. Storage

Capacity for storage is provided in Area 247 in the Fuel Building (see Figure 1.2-38). Wastes (e.g., spent resins and filters, dry active waste, waste oil) may also be stored pending shipment outside the Fuel Building or in a warehouse in a fenced area designated for radioactive materials staging and handling. This fenced area is part of the Radiologically Controlled Area. For higher radiation level waste containers, appropriate shielded storage containers are provided to ensure that radiation levels at and beyond the radioactive materials staging and handling area fence are within 10CFR20 limits for unrestricted areas. Storage of radioactive materials (e.g., contaminated scaffolding and outage equipment, etc.) may also be stored in these areas.

Storage time is a variable and depends on shipment schedules and disposal site availability. Interim storage of low-level radioactive waste may be required due to disposal site unavailability. Interim storage locations include a warehouse area, (2K7) and a fenced area (3L15) shown in Figure 1.2-1. Interim storage areas would be part of a Radiologically Controlled Area.

Prior to container shipment, drum smear samples are taken to determine the surface contamination. If required, the container surface is decontaminated and smear samples are taken again to determine if the desired decontamination has been achieved.

3. Shipment

Radwastes are stored in a designated staging area prior to shipment. Shipment of the radwaste originates from the staging area. All radwaste shipments will be in compliance with the applicable regulatory standards and requirements of the NRC, DOT, Texas Regulations for Radiation Control, and burial site.

11.4.2.8 Instrumentation and Control

The instrumentation and controls of the SWMS & WBS are designed by the equipment supplier.

11.4.2.9 Safety Evaluation

The SWMS & WBS are not safety-related and cannot affect the safe shutdown of the plant or the operation of other systems which are required to safely shut down the plant. Failures of the

SWMS & WBS have been analyzed. The thyroid and whole body doses are below the values set forth in 10 CFR 100.

11.4.2.10 The Old Steam Generator Storage Facility

The Old Steam Generator Storage Facility provides a secure long-term storage facility for the four Unit 1 Old Steam Generators and the Reactor Head Vessel Heads and associated Control Rod Drive Mechanisms for Units 1 and 2. Design features for storage of radioactive materials in the Old Steam Generator Storage facility are described in Section 12.2.1.6.

TABLE 11.4-1
SOURCE INPUTS[a] (2-UNIT OPERATION)

Type of Waste	Expected Annual Disposal Volume (ft^3)	Avg. Container Contact Dose Rate
Dry Active Waste (DAW)	10,000	≤ 50 mR/hr.
Processed Wet Waste		
Evap Conc and Chem Drains	270	≤ 200 mR/hr.
Filter Cartridges	1,000	≤ 5 R/hr.
Resins	1,510	≤ 100 R/hr.
Total Waste	12,780	

a) Based on current industry average for PWRs during the mid 1980's. Actual disposal volumes are dependent on waste processing and volume reduction techniques used.

TABLE 11.4-2
COMPONENT DESIGN PARAMETERS

(DELETED)

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

11.5.1 DESIGN BASES

The Process Radiation Monitoring System (PRMS) provides a means for measuring and controlling radioactive process streams and effluents throughout the plant. The system is in compliance with the requirements of 10 CFR Part 50, 10 CFR Part 20, NRC Regulatory Guide 1.21, and GDC 60, 63, and 64. The system aids in ensuring the protection of the general public and plant personnel from exposure to radiation or radioactive materials in excess of those allowed by applicable regulations or governmental agencies. In addition, the system is designed in response to 10 CFR Part 50, Appendix I, to limit radiation levels to ALARA. The design objectives of this system for normal operation are as follows:

1. To provide continuous indication and on demand record of radiation levels, and automatic record of alarms in process and effluent streams over the range from clean-plant background to levels commensurate with Technical Specification limits.

2. To provide data useful in reporting total released activity.

3. To give early warning of increasing levels of radioactivity in process and effluent streams by using an alert-alarm set point.

4. To give an alarm and initiate a control function, where appropriate, upon high radiation levels in plant process and effluent streams using a high-alarm set point.

5. To give an indication that there is a monitor malfunction, equipment failure, or loss of counts by using a channel signal valid data check. If the test is negative, a “fault” alarm is initiated.

The system also performs post-accident functions as described in Sections 6.2.4 and 7.5 to satisfy GDC-13 and 54.

11.5.1.1 Digital Radiation Monitoring System

The Digital Radiation Monitoring System (DRMS) is comprised of the following subsystems:

1. The Area Radiation Monitoring System (ARMS), which continually monitors radiation fields in various representative regions within the plant. The ARMS is described in Section 12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation.

2. The PRMS, which provides a means for assessing radioactivity levels in plant process and effluent streams, and controls plant process and effluent streams including the handling and processing of radioactive waste.

11.5.1.2 Process Radiation Monitoring System

The PRMS is comprised of the following monitors:

1. Airborne, Steam and Gas Process and Effluent Radiation Monitors (also see Section 12.3.4)

    a. Containment air monitors (1 per unit)

    b. Control Room ventilation intake monitors (4, 2/intake)

    c. Main steam and feedwater area vent duct monitors (1 per unit)

    d. Auxiliary Building vent duct monitor (1, common)

    e. HVAC room vent duct monitor (1, common)

    f. Fuel Building vent duct monitor (1, common)

    g. Safeguard Building vent duct monitors (1 per unit)

    h. Plant vent stack monitors (2/stack, common)

    i. Main Steam Line Monitors (4 per unit)

    j. Steam generator Leak Rate Monitors (4 per unit)

2. Liquid Process and Effluent Radiation Monitors

    a. Deleted

    b. Deleted

    c. Component cooling water monitors (3 per unit)

    d. Service water monitors (2 per unit)

    e. Deleted

    f. Steam generator blowdown sample monitors (1 per unit)

    g. Deleted

    h. Turbine Building sump effluent monitors (1 per unit)

    i. Liquid waste effluent monitor (1, common)

    j. Waste gas monitor (1, common)

    k. Condenser off-gas monitors (1 per unit)

l. Deleted

m. Failed-fuel monitors (1 per unit)

n. Auxiliary Building to LVW Pond monitor (1, common)

These monitors, as described in the following sections, serve in conjunction with varied portable and mobile monitoring equipment as well as with a comprehensive sampling program. Since continuous on-line isotopic analysis and measurement of exceedingly low radionuclide concentrations are not within the practicable state of the art, the sampling program is the primary method for quantitatively and qualitatively evaluating process system and effluent activity levels.

In the event of a radiation release the PRMS and the ARMS can provide immediate information about the concentration and dispersion of radioactivity throughout the plant, enabling operating personnel to evaluate the severity and to mitigate the consequences. For some anticipated operational occurrences resulting primarily from operator error and under certain conditions (where radiation levels exceed set points, the Radiation Monitoring System (RMS) automatically actuates necessary valves and thereby limits the consequences of the release.

Additionally, process monitors are provided upstream of potential radionuclide discharge points between the potential sources and release points as described in this section and as shown on flow diagrams.

The effluent release paths are monitored by the following:

The Containment air monitor provides for monitoring of a continuous sample drawn from the Containment for noble radioactive gas, iodines in elemental and organic-compound forms, and activity in particulate form. During purges or vents, Containment effluent is discharged via the plant vent stacks where it is monitored by off-line particulate, iodine, and gas (PIG) monitors and/or the Wide Range Gas Monitors (WRGM).

The condenser off-gas and waste gas effluent are monitored by beta-sensitive detectors enroute to the primary plant ventilation system and plant vent stack where they are ultimately monitored and discharged.

When the Liquid Waste Processing System is in the proper valve configuration, liquid can be discharged to the Circulating Water System and will be monitored by a gamma-sensitive detector before release to the circulating water discharge tunnel. Turbine Building sump effluent is monitored by gamma-sensitive monitors prior to release to the Low Volume Waste Pond. Upon detection of radioactivity, this effluent is diverted to the Waste Water Holdup Tanks where it is sampled and if appropriate, released from the holdup tanks on a batch basis to the Circulating Water Discharge Tunnel.

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 Design Criteria - Continuous Monitoring

The following criteria govern the design of the PRMS:

1. To facilitate compliance with applicable regulations (e.g., 10 CFR Part 20), monitors and detectors have sensitivities and ranges compatible with radiation levels anticipated at specific detector locations.

2. The alarm setpoints and other important database parameters are controlled by approved plant procedures utilizing designed supervisory control features inherent to the process monitors.

3. The detected radiation levels of the monitored process and effluent systems and the most recent 28 daily, 24 hourly, or 24 10-minute averages are available on the RMS consoles in the Control Room. This information may also be printed on demand on printers located in the Control Room.

4. Control Room alarms annunciate high radiation levels and radiation monitor malfunction, as appropriate.

5. Monitor electronics have independent power supplies.

6. A check source or current is provided for each detector, which is operable from the Control Room, except the failed fuel monitor which is checked during preventive maintenance.

7. Local alarms and indication are provided for monitors where local process control panels are provided. (See Table 11.5-3.)

8. Environmental design conditions for the components are as described in Table 11.5-2.

9. Monitors are placed where they are accessible for maintenance and inspection.

10. Monitors are placed where the background is determined to be the lowest available, in accordance with governing ALARA design considerations, and for detection sensitivity considerations.

11. Detectors, electronics, and electric equipment associated with the containment air and Control Room ventilation intake monitors are seismically qualified.

Sufficient lead shielding against ambient background radiation is provided so that adequate sensitivity is achieved. The count rate (or detector output signal) at the minimum detectable activity (MDA) will be equal to or greater than twice the design background standard deviation. Design background for the sensitivity limit is considered to be the larger of either 1 mR/hr of 1 MeV gamma radiation or a reasonable design maximum for the area in which the monitor is located.

The high-alarm set points are at a value not exceeding plant technical specification limits, assuming the maximum anticipated flow rates.

The proper operation of some detectors are checked with a built-in check source which can be controlled from the Control Room. Check source indications are dependent upon expected background and are engineered to yield a positive indication under all anticipated conditions. The design of air stream sampling considers the guidance of ANSI N13.1-1969. The type of monitors used are off-line, on-line and in- line, as shown in Table 11.5-1. Use of in-line monitors provides complete monitoring of the sampling media and reduces sampling errors from auxiliary devices, pumps, and valves.

The off-line monitors are provided, when required, because of temperature, line size, or other considerations. Where a pressure differential between sample connections is insufficient to ensure proper off-line flow, a pump with flow indicator is provided. All off-line sample lines have isolation valves which allow removing the monitor from service. The off-line monitor sample chamber and auxiliary piping is made of stainless steel of the quality necessary to conform to ASTM standards.

An on-line (or adjacent-to-line) type monitor is provided because of the special considerations of the monitored fluid. The on-line monitor has the advantage of simplicity, freedom from sampling problems, and flow control in that existing piping need not be altered, since flow impediments and sediment collection points are avoided. The use of on-line monitors is based on sensitivity considerations.

11.5.2.2 Detector Location Selection

Detector locations conform with the general criterion of NRC Regulatory Guide 1.21, which states that “all normal and potential paths for release of radioactive material during normal reactor operation including anticipated operational occurrences should be monitored.” In accordance with this guide, the points to be monitored are those that provide data on effluent releases to the plant environs; these monitoring points are indicated schematically on diagrams referenced in Table 11.5-1. Monitors are provided for the following:

1. Process streams that normally discharge low-level activity directly to the environment.

2. Continuous process streams that discharge directly to the environment but do not normally carry radioactive material. Such monitoring indicates if any radioactive leaks into these process lines have occurred.

3. Process lines which contain radioactivity but do not normally discharge to the environment. Such monitoring indicates if process malfunctions have occurred.

11.5.2.3 Expected Radioactivity Concentrations and Quantity Measured

The expected radiation levels in the process and effluent streams are such that concentrations in gaseous effluents at the Exclusion Area Boundary and liquid effluents at the discharge point are a small fraction of 10 CFR Part 20 limits. The calculation of radioactive concentrations and the levels to be monitored or sampled are described in Section 11.1 and Table 11.5-4.

Each channel, except the iodine monitors, measures gross radioactivity.

11.5.2.4 Detector Type, Sensitivity, and Range

The range of each detector, along with other pertinent information such as detector type and reference nuclide are summarized in Table 11.5-1.

Detector location and sample line routes from sampling point to detector are chosen to minimize sample line length, and the number of direction changes are chosen to minimize transport losses. Sample lines are appropriately sized; where applicable, stainless steel tubing is made with long radii elbows in conformance with ANSI N13.1-1969.

Process and effluent stream activity detectors are scintillation crystals, Cd Te (Cl), PM Tubes, GM Tubes, etc. which detect beta or gamma radiation over an energy range appropriate for the process stream. Design sensitivities and ranges have been selected so that the instrument either reads on scale or stays within system limits during normal operation, taking into consideration the limitations of such equipment.

11.5.2.5 Digital System Description

11.5.2.5.1 General

The RMS is a dedicated distributed microprocessor based digital monitoring system. System communication connections require only two twisted-shielded pairs of low-voltage cable. Details are provided in the following text and Figure 11.5-1. Certain channels are seismically qualified to remain operable during and after the SSE. The Control Room Cabinet consist of Liquid Crystal Displays (LCDs) and printers. The monitor detector locations have control, data processing, data storage, and multilevel alarming features. See Table 11.5-3.

The microprocessor-based RMS is comprised of the following components:

1. Display/Operator (SCADA-A & B), color LCDs, alarm printers, and keyboards; two operator console LCDs (View Nodes) and keyboards

2. Dedicated microprocessors for each monitor

3. Remote displays and controls on process system operator panels as applicable

4. Monitors with associated equipment

5. Mirror SCADA-A & B with dedicated LAN to support the LAN-based radiation monitoring clients.

6. Control Room equipment racks (two) furnished with display/control modules for selected monitors

The system concept is a distributed data base with each individual monitor processor maintaining its own data base and stored data. A stand-alone configuration includes redundant computers in a Display/Operator Cabinet that handle five loops of up to 31 monitors per loop. Alarm messages are sent to the Control Room from the detector microprocessor when polled by the respective PC-11. The requested data are then returned to the PC-11 where they are displayed, printed, and announced. Alarms are displayed. Operator-initiated display requests cause

requests for data from the monitor processors. The computers contain the communications links to the monitor processor loops and the necessary memory and programming to provide overall system status, monitor group profiles, and individual monitor trend displays. These computers are interconnected to provide alternate communications paths, thus tolerating a single communications cable fault or monitor malfunction without loss of function in the system. In addition to the display requests from the keyboard, other control functions such as check-source readings, pump motor control, and purge control are provided.

Certain radiation monitor channels are seismically qualified, because of their importance to the safe operation of the plant. These channels are capable of withstanding a Safe Shutdown Earthquake (SSE). Dedicated control/display modules on seismically qualified cabinets for the containment air monitors, plant vent stack monitors (WRGMs and PIGs), and the class 1E Control Room ventilation intake monitors are provided in the control room. The individual control/display modules in the cabinets utilize the same microprocessor chip as the monitor processor but on a simpler board. These modules perform basically the same control and display functions as the PC-11A, B LCD Displays, but with limited display capability. The control/display module has lighted alarm indicators, pushbuttons for the check-source test, purge or filter step, and pump motor control. Since the seismic channels are also connected to the PC-11A, B Displays, the normal use of these modules is as a backup to the PC-11A, B Displays which are not seismically qualified.

Digital communication is the basis for component interface for the multiprocessor system. Communication circuits tie the monitor processors and the Control Room Display/Operator Cabinet into four loops. The communications cabling uses two twisted-shielded pairs in a daisy-chain configuration. Each daisy chain forms a loop from Scada A through a group of monitors to Scada B inside the Control Room. It provides two communication paths to and from the Control Room. In case of a failure in a loop section the monitor processors on the first console side of such a failure communicate with that console, and those on the other side communicate with the second console. The consoles share information with each other and lose no information after any single communication failure. Communication consists of request for information via alarm and data polls or commends from the Control Room to the field microprocessor units.

Each monitor processor has a data base which includes count rate, conversion factors, high alarm limit, alert alarm limit, check-source limit, and where monitored, sample and process parameters. Calculations are performed by microprocessor using these parameters and the raw data from the detector. The raw data from the detector are accumulated into two separate computational sections. One section performs averaging calculations and history filing; the other computes the current rate and tests for alarm conditions.

Processed data are averaged and stored in memory for a historical trend. The history file of detector data consists of 24 10-min averages, 24 1-hr averages, and 28-1 day averages.

Alarm determinations are based on having a significant number of counts available to calculate, with a 95 percent confidence level that count alarm rates are being reached. Counts are incremented into an accumulator and tested for significance at time intervals consistent with system alarm response requirements. If the count is significant and the rate is beyond an alarm limit, alarming occurs. The rate computed in this alarm section is displayed as the current rate.

11.5.2.5.2 Monitoring Assemblies

Each monitoring assembly is the equivalent of a digital ratemeter. Each Scada with its color LCD display, keyboard, and printer is a remote digital display and recorder for the monitoring assemblies. All field wiring for communication circuits consists of twisted-shielded pairs that are daisy-chained from monitor to monitor and to the display cabinet. Each monitor and the display cabinet are furnished with safety related or reliable non-safety related power (backed by diesel/ non-safety related station batteries) as applicable.

At each monitor, control, data processing, data storage, and multilevel alarming are all performed by the local microprocessor independently from the rest of the system. If Control Room displays and computers are down, the data stored locally at each individual monitor are available for later communication when the Control Room displays and computers are back on line. The hourly and daily averages continue to be updated and stored 24 hours may pass without loss of hourly averages. Twenty-eight days may pass without loss of daily averages. No data need be lost as a consequence of multiple failures in the equipment common to all monitor channels.

The microprocessor at each radiation monitor assembly controls the monitor, processes and stores data, and communicates messages through intervening monitors to the Control Room Display/Operator Cabinet.

The parameters (such as setpoints, conversion factors and channel identification) that specify how to control each channel and how to process its data are provided in the microprocessor database.

Data processing at each monitor microprocessor is briefly listed as follows:

1. Monitor Microprocessor Controls (on command from the Control Room Display/Operator Cabinet)

    a. Operates the check source upon manual command from the Control Room Display/Operator Cabinet

    b. Steps the filter (where applicable) upon manual command from the Control Room Display/Operator Cabinet

2. Monitor Microprocessor Data Processing

    a. Accumulates gross pulses from the detector up to a maximum pulse rate of 107 cpm

    b. Subtracts the operator entered background value from the gross count

    c. Converts net count rate to engineering units ($\mu Ci/cm^3$) via a specified conversion factor

    d. Where appropriate, via a parameter input point, measures the process fluid flow rate analog signal and uses it to convert $\mu Ci/cm^3$ to $\mu Ci/sec$, using a specified scale factor

e. Differentiates the filter-integrated particulate monitor data from fixed particulate monitor data

f. Tests for alert and high alarms on the basis of a significant quantity of counts accumulated. Actuates applicable audio and visual alarms and control relays. Resets the local horn. Issues a status message to the Control Room Display/ Operator Cabinet when the alarm status changes.

g. At the appropriate stage of processing, tests for the other alarm conditions listed previously

h. Checks for data validity (see Subsection 11.5.2.5.1) and marks each stored value questionable if the majority of its counts are not valid

i. Maintains a rotating file of past values of radiation data: 24 10-min averages, 24 1-hr averages, and 28 1-day averages.

j. Uses a specification table (Database items such as conversion factor, etc.) to control the processing. All entries for a monitor's specifications are sent as messages to that monitor's microprocessor, where they are stored in the specification table.

11.5.2.5.3 Control Room Display/Operator Cabinet

The PC based system is comprised of two operator consoles and a Display/Operator Cabinet. It is a complete, stand-alone system, as shown on Figure 11.5-1. The RMS consoles are dedicated to radiation monitoring and function independently of the plant computer.

The central (RMS) Display/Operator Cabinet (PC-11 SCADA - A & B, see Figure 11.5-1) is divided, with each section having a 18-in. color LCD display, a keyboard, a printer, and a computer.

The operator consoles (PC-11 View Nodes - A & B, see Figure 11.5-1) each have a 18-in. LCD display and keyboard which are wired to a dedicated PC at the Display/Operator Cabinet.

The consoles give the operators consolidated and fully processed information on the RMS throughout the plant. The basic displays on a RMS display are six grids and a control menu display. The characters identify the monitor and the background color indicates monitor status,(i.e. alarm conditions, check source testing, off scan, no response, and normal operation). Additional displays/functions for a radiation monitor/channel are accessed from a standard keyboard.

The next level of displays allow operators to request detailed information on individual monitors. The following are the baseline displays/functions:

1. Monitor Specific Display

   The monitor specific displays contain information such as the current radiation reading, real time trends, alarm setpoints, database item, communication status, etc. For monitors

with flow, a diagram showing pumps, detectors, and other prominent components is also provided.

2. Monitor Trend Display

   The monitor trend display is used to access the trend histories of 24 10-min. averages, 24 1-hour averages, or 28 1-day averages. These trends are displayed in bar chart format scaled to the high alarm limits. The monitor trend display also provides information such as the current activity in engineering units, the data base background value, the high alarm setpoints, and process and sample flow rates, if applicable. In addition to this, a real time trends screen for activity, sample flow, and process flow, as applicable, is provided.

3. Controls Display

   The Control Displays provide cursor selectable control options for pump on/off, purge on/off, filter advance, and check source requests. Supervisory control display allows alteration of database and alarm setpoints.

4. Monitor/Channel Data Base Items Display

   The monitor/channel data base items display provides the current values for the monitor/channel data base items for a particular monitor/channel. The data base changes are password controlled.

5. Alarm History Display

   The alarm history display lists messages regarding the current status of the Radiation Monitoring System.

6. Data Base Compare

   PC-11 provides an operator selectable function to compare the individual Radiation Monitor data base with a resident “master” data base for proper configuration control. This option prints a report identifying any data base discrepancies.

The four computers communicate with each other and with the remote monitor assemblies.

11.5.2.5.4 Control Room Equipment Racks

A seismically qualified control/display module for each Containment Air, plant vent stack (WRGMs and PIGs) and Control Room Ventilation intake monitor is located within seismic equipment racks in the Control Room. Digital displays are provided for each channel, in addition to switches for pump control, filter step, check source operation, and alarm acknowledge. The channel status indicators (operate, alert, high) are indicated by means of green, yellow, and red lamps, respectively, provided on each module.

Each seismic channel has a separate radiation detector with internal power supply. A local microprocessor is provided for each monitor. Each microprocessor has three communication

ports which provide for connection to the central Display/Operator Cabinet and the control/display modules in the Control Room.

Should communications between local microprocessors and the Control Room fail, the microprocessor at the local detector would continue to monitor radiation levels, and to store daily averages for up to 28 days. Thus, a high degree of reliability and independence is assured. The microprocessor is also equipped with a connector where a portable digital readout device can be connected to display data. The monitoring system has independent power supply electronics for each monitor.

Non-seismic monitors are locally identical to seismic monitors, but are provided with necessary communication ports to communicate with the non-seismic Display/Operator Cabinet computers in the Control Room. Therefore, the non-seismic channels are inherently as reliable as the seismic channels, except for possible loss of Control Room readouts in the event of an SSE.

11.5.2.5.5 Set Points

Set points for appropriate process and effluent monitors are established to meet plant Technical Specifications and Radiological Effluent Controls Program limits, which are governed by 10 CFR Parts 20 and 50, Appendix I, objectives. Set points for process monitors are established to provide timely warning of increased system activity that requires corrective action to maintain safe operating conditions.

Two independently adjustable bi-level radiation set points are provided. The alert set point activates only an alarm, while the high set point not only is alarmed but also initiates control action where appropriate. Alert set points may be set as close to the background as feasible to provide early warning without generating statistically nonsignificant alarms. The bases for alarm set points and the associated flow diagram references are listed in Table 11.5-1. Alert-alarm set points are between background and the high-alarm set point. The system is designed to provide physical security for access to alarm setpoints. These setpoints are administratively controlled by approved station procedures.

The instrumentation and/or control diagrams for safety systems shown/identified in Section 1.7 and 7.2 include radiation monitoring systems.

11.5.2.5.6 Annunciators and Alarms

All process and effluent radiation monitors are annunciated on the RMS Display/Operator Cabinet in the Control Room. The individual detector microprocessors respond to pollings from the respective PC-11 by initiating alarm messages in the control room when set points are exceeded, although all automatic control functions and local alarms are initiated directly from the microprocessor. The routine and automatic polling of monitors is for the purpose of accumulating data for trend history, radiation alarms and monitor failures. The functions of alarming via a LCD Display, sounding a horn, and printing alarm events on a printer are performed by the system.

The RMS Display/Operator Cabinet stores complete alarm status for all radiation monitors. Keyboard access is provided for alarm status and alarm sort screens.

Each new alarm condition is logged at redundant alarm printers. The alarm message, as typed, contains the date, time, alarm code, channel or monitor identification, and brief description. When channels in alarm return to normal, this condition is logged in the same manner.

11.5.2.6 Airborne and Steam Radioactivity Monitoring Systems

Radiation monitors are provided for continuous detection and measurement of airborne radioactivity for gaseous process streams and for plant gaseous effluents as summarized in Table 11.5-1. General design and performance objectives for all elements of the PRMS are covered in Subsections 11.5.1 and 11.5.2. In addition, the following criteria for air sampling are met:

1. Detectors are located as close to sampler intakes as feasible.

2. In the design of sample nozzles and lines, the guidelines of ANSI N13.1-1969 are considered to obtain representative sampling and reduce settling and plateout losses of particulates in transport to the detector.

3. The Containment atmosphere particulate and gaseous radioactivity monitors are part of the reactor coolant pressure boundary (RCPB) leak detection system. Accordingly, all elements of these channels required for radiation indication are designed and qualified to remain functional following an SSE, in compliance with NRC Regulatory Guide 1.45.

11.5.2.6.1 Sampling Devices

For each off-line gaseous monitor, a sample is drawn from a system through a sample line to the monitor skid. The sample is routed through the monitor skid and then returned to the system from which it was extracted. Samples from the plant vent stacks are drawn using isokinetic nozzles.

Sample pumps are utilized to draw an appropriate sample through the monitor. Each monitor has a low sample flow alarm. A local flow indicator is provided for vent stack monitors which have particulate and iodine filters, so that the total volume that has passed through the filters can be determined. The filter papers used to collect particulates have a collection efficiency of at least 99 percent for 0.3-micrometer particulates. The cartridges used to collect iodine have been shown to have a minimum efficiency of 95 percent for elemental and organic iodine.

Each off-line gas monitor has manually operated sample valves. This allows room air to be routed through the gas monitor to check the background radiation level and allows for samples to be taken or for the introduction of calibrated gas to check monitor calibration.

The location of sample probes and off-line monitors has been chosen to minimize sample plateout. Unavoidable bends are made with radii not less than five times the tubing diameter. Stainless steel lines and appropriate sampling valves are used. In addition, a local sample tap is provided for the containment air monitoring systems so that a test can be performed to determine the sample deposition (plateout) after system installation.

11.5.2.6.2 Containment Air Monitoring System

This system is provided to monitor Containment air continuously for particulate, iodine (elemental, particulate, and organic forms), and noble gas activities. The monitoring system takes a continuous air sample from the Containment atmosphere, which is drawn outside the Containment in a closed system. The sample air passes through a particulate filter which is continuously monitored by a beta-sensitive scintillation detector. The filter paper collects 99 percent of all particulate matter greater than 0.3 micrometers in size. Shielding is provided to reduce ambient background radiation to a level that provides adequate detector sensitivity. After the sample leaves the particulate monitor, it passes through a closed system to an adsorber cartridge to collect iodine, which is viewed by a shielded gamma-sensitive scintillation detector. The particulate filter and iodine filter are replaced as required to support operation of the monitor. When the sample air leaves the iodine cartridge it is directed through a closed system to a shielded stainless steel gas sampling chamber viewed by a beta-sensitive scintillation detector. The sampled air is finally returned to the Containment atmosphere. Indication and annunciation are provided in the Control Room. The detection of high radiation levels by the particulate or noble gas channels causes the high-level set point to trip and initiates Containment Ventilation Isolation.

The particulate and gas monitor are also used as part of the RCPB leakage detection system. The sensitivity and response time of this part of the leakage detection system, which is used for monitoring unidentified leakage to the Containment, is discussed in section 5.2.5.3.

The containment atmosphere can also be manually sampled in a number of ways and via a number of outlets if required. Some examples are: by removal of a section of the existing sampling line particulate filter after a known collection period and flow rate and submitting the filter for laboratory assay; by means of an in-line filter and vacuum pump sampling from the test valve in the containment sampling line; and by evacuated flask sampling from the same test valve. Other sampling points potentially exist on the continuous air monitor unit as well as from other containment penetrating equipment, such as hydrogen analyzer equipment.

11.5.2.6.3 Plant Vent Stack Monitoring System

Each plant vent stack is equipped with a gas monitor. They are off-line monitors that sample the plant vent effluent prior to discharge.

The plant vent stack gas monitors draw representative air samples from the plant vent stack via isokinetic nozzles in the stack. Lead shielding is provided to reduce background to levels that allow the detection of the specified instrument ranges given in Table 11.5-1.

For normal range noble gas monitoring when no automatic control functions are required, the Plant Vent Stack Gas Monitor may be used as a backup for the Wide Range Gas Monitor, which is the primary monitor that is used for plant vent stack noble gas detection.

A separate Wide Range Gas Monitor (WRGM) is provided for each stack for extended range noble gas monitoring. Each WRGM has two isokinetic nozzles; one for sampling during normal conditions and one for accident conditions. The stack sample passes first through a sample conditioning unit which filters particulates and iodine and may be used to take grab samples. The WRGM particulate and iodine filters are replaced at least weekly and counted for activity for release accountability. The sample then passes through the detector assembly which uses three

detectors to cover the complete range required. These Wide Range Gas Monitors satisfy the requirements of NUREG-0737, II.F.1 for provisions for sampling plant effluents for iodines and particulates and for noble gas effluent monitors for the plant vent.

The detection of high radiation levels by the WRGM causes the high-level set point alarm to initiate the automatic closure of the gas release valve in the waste gas processing system.

11.5.2.6.4 Control Room Ventilation Intake Monitors

These monitors continuously assess the intake air to the Control Room for an indication of abnormal atmospheric radiation concentration. Two monitor assemblies per intake are provided and are powered from separate electrical supplies. In the event of high radiation, the monitors initiate Control Room ventilation emergency recirculation through filters.

11.5.2.6.5 Condenser Off-Gas Monitor

Gaseous samples are drawn through an off-line system by a vacuum pump from the discharge of the vacuum pump exhaust header of the condenser. This channel monitors the gaseous sample for radioactivity which would be indicative of reactor coolant leakage to the secondary side. The gaseous radioactivity levels are monitored by a beta-sensitive scintillation detector. From the monitor the condenser off-gas is then routed to the Primary Plant Ventilation System. Indication and annunciation are provided in the Control Room. Each of the off-line monitor assemblies comprises a thin plastic scintillator, photomultiplier tube, four pi lead shielding, Cl-36 check source, indicators, and associated electronics. This monitor complements the steam generator blowdown process sample monitors. Monitor range is discussed in Table 11.5-1, and expected concentrations are discussed in Section 11.1.

The COG Radiation Monitors have no control functions. They are installed to aid in detecting a slow propagating steam generator tube leak (detection of tube rupture). Note that argon gas may be injected into the RCS to raise the primary coolant activity. This helps to ensure a reliable COG Monitor response in the absence of other fission and activation product inventories.

11.5.2.6.6 Waste Gas Monitor

This in-line monitor provides an indication of magnitude of the gross gaseous activity in the GWPS by monitoring the input line to the GWPS compressors by means of a CdTe(Cl) scintillation detector. Indication and annunciation are provided in the Control Room, as well as on the local process control panel.

11.5.2.6.7 Auxiliary Building Vent Duct Monitor

This monitor assembly is mounted directly in the ventilation duct with the detector window looking at the duct gas. The beta-gas radiation detector is a thin plastic scintillation phosphor behind a thin aluminum window mounted integrally with a photomultiplier tube. A Cl-36 beta check source is provided for verification of channel operation. This channel monitors the accumulation of airborne radioactivity that is drawn by the Auxiliary Building ventilation system from the various equipment rooms, corridors, and cubicles of the Auxiliary Building. This monitor also initiates the automatic closure of the gas release valve in the waste gas processing system on detection of high radiation. Indication and annunciation are provided in the Control Room.

11.5.2.6.8 HVAC Room Vent Duct Monitor

This monitor is identical to the beta-gas scintillator of the Auxiliary Building vent duct monitor. It is located inside the heating and ventilation room ventilating duct to monitor potential airborne radioactivity in the ventilation system and to give an indication of possible changes in process systems.

11.5.2.6.9 Fuel Building Vent Duct Monitor

The Fuel Building vent duct monitor is identical to the monitor described in Subsection 11.5.2.6.8, except that it provides an indication of Fuel Building airborne activity. Fuel Building radiation monitors are not safety-related; however, an appropriate operational Quality Assurance program will be applied.

11.5.2.6.10 Safeguard Building Vent Duct Monitor

These monitors are similar to the monitor described in Subsection 11.5.2.6.8, except that they provide an indication of Safeguard Building airborne activity.

11.5.2.6.11 Main Steam and Feedwater Area Vent Duct Monitor

These monitors are similar to the monitor described in Subsection 11.5.2.6.8, except that they provide an indication of airborne activity in the main steam and feedwater penetration area.

11.5.2.6.12 Main Steam Line Monitors

These monitors utilize GM tube detectors to monitor increases in secondary side activities. They are mounted adjacent to the main steam lines and are located between the PORV and the first safety valve.

11.5.2.6.13 Steam Generator Leak Rate Monitors (SGLRM)

These monitors are provided to detect N-16 radioactivity in the main steam lines. The SGLRMs are mounted on the main steam lines just upstream of the MSIVs. Detected radioactivity is indicative of a primary-to-secondary system leak and provides an estimate of the range of potential steam generator leak rate. Because these channels monitor the secondary system radioactivity levels, it supplements the main steam line monitors, the steam generator blowdown sample monitor and the condenser off-gas monitor. Gamma-sensitive scintillation detectors are provided. Monitor output indication and alarm annunciation are provided at the Control Room RMS Display/Operator Cabinet.

The SGLRMs have no control function. They are installed to aid in detecting a slow propagating steam generator tube leak (steam generator tube rupture). The primary-to-secondary leak rate program at CPNPP uses the SGLRMs as the primary source for indication and the condenser off gas monitors are used as a back-up/confirmatory indication.

11.5.2.7 Description of Liquid Monitors

Each liquid monitor channel of the system contains an integrated modular assembly; specific details of each monitor are described in Subsections 11.5.2.7.1 through 11.5.2.7.12.

1. General Description - Sampling Devices

There are four basic types of liquid monitors used in the system:

- Off-line without a sample pump; (Sodium Iodide (NaI(Tl) Detector)
- Off-line with a sample pump; (Sodium Iodide (NaI(Tl) Detector)
- Off-line failed fuel; (GM Tube Detector)
- Adjacent-to-line; (Sodium Iodide (NaI(Tl) Detector)

Off-line monitors draw a sample from a process line, through a shielded sample chamber, through the sample pump, where one is required, and then return it to the process system. Each sample pump is capable of drawing at least one gallon per minute of liquid through the monitor. The sample flow rate is controlled by means of a manual valve. All off-line monitors have a low-sample-flow alarm. Adjacent-to-line monitors provide continuous monitoring of the process fluid through the process pipe line. The detector shield configuration is factory designed to accommodate mounting installation on the existing plant piping. A flow switch is not required because the detector is mounted adjacent-to-line with the pipe.

The monitor inlet and outlet lines are flanged and the sample piping has isolation valves so that the monitor can be disassembled for decontamination.

2. Detector-Photomultiplier-Preamplifier Unit

With the exception of the GM Tube, each detector is a NaI gamma-sensitive scintillation detector. A photomultiplier and preamplifier are mounted integrally with the detector. The detectors are designed to remain fully operational over a wide range of temperatures, as shown in Table 11.5-2. If they are exposed to high radiation transients up to 100 times the channel range, the channel maintains its operation and returns to normal functioning when the transients have subsided. Since gamma detectors are used, it is possible to compare the monitor readout with the results of grab samples analyzed in the plant multichannel analyzer as a check of proper monitor operation. Solenoid-operated check sources are provided to check detector response. Monitor characteristics are listed in Tables 11.5-1 and 11.5-3. Monitors with local process control panels have local radiation alarms and/or indication of channel radioactivity levels.

11.5.2.7.1 Steam Generator Blowdown Sample Monitor

This channel monitors the liquid phase of the secondary side of the steam generators for radioactivity to provide an indication of a primary-to-secondary system leak. This provides backup information to the condenser off-gas monitor. The radiation detector is able to monitor radioactivity from blowdown for each steam generator. Blowdown sample lines from each of the steam generators are combined in a common header, and the common sample is continuously monitored by a shielded gamma-sensitive detector located downstream of the sampler assembly. After sampling and monitoring, the blowdown sample is routed to the atmospheric drain tank for return to the Condensate System. Control Room alarm and indication are provided to alert operating personnel. Additionally, an alert alarm is also provided on the local process panel. In

the event of high activity, the monitors high-level set point trips and initiates an alarm and the automatic closure of the isolation valves in the blowdown and sample lines. Prior to this event, the alert alarm set point gives an alarm and indication of increasing activity. The isolation valves for blowdown sampling can be opened by overriding the monitor, as needed, for operation and analysis. Identification of which steam generator is leaking is made by overriding each of the sample line isolation valves and drawing samples from each steam generator blowdown line separately for analysis.

11.5.2.7.2 Steam Generator Blowdown Monitor

“The Steam Generator Blowdown Monitor has been removed”.

11.5.2.7.3 Liquid Waste Monitor

Discharges from the LWPS (see Section 11.2) are continuously monitored by a shielded gamma-sensitive (NaI) scintillation detector. A radiation monitor and a control valve discharges processed waste to the circulating water discharge tunnel. The discharge control valve is administratively controlled; if activity concentrations exceed the discharge monitor high radiation alarm set point, automatic closure of the discharge control valve is initiated. Indication and annunciation are provided on the WPS control panel and in the Control Room.

11.5.2.7.4 Auxiliary Steam Condensate Monitor

This channel has been removed from the plant. Sample fluid is drawn from the WPS waste evaporator, floor drain waste evaporators and the BRS recycle evaporator auxiliary steam condensate return lines prior to delivery to the auxiliary steam drain tank, and analyzed periodically whenever the evaporator packages are operated. Upon detection of predetermined levels of radioactivity, manual closure of isolation valves is initiated to minimize contamination of systems serviced by the auxiliary steam system. A heat exchanger is located upstream from the radiation monitor in the process line to cool the sample.

11.5.2.7.5 Service Water Monitor

These monitors are provided to monitor the Service Water Systems for radiation since in leakage from radioactive fluid systems could cause potential radioactive leakage to the environment. A shielded NaI scintillation detector is located in an off-line sampler assembly downstream of each component cooling water heat exchanger to monitor service water being discharged. Indication and annunciation are provided at the Control Room RMS Display/Operator Cabinet.

11.5.2.7.6 Component Cooling Water Monitor

This off-line channel monitors the CCWS for radiation. Any radiation detected would be indicative of a leak of reactor coolant from the RCS or from the Residual Heat Removal (RHR) System to the CCWS.

A monitor sample line is located in each of the component cooling water return headers upstream from the surge tank. A shielded NaI gamma-sensitive scintillation detector monitors the pump-drawn fluid which is returned to the CCWS after monitoring. If a high radiation level is detected, an alarm signal is initiated in the Control Room.

11.5.2.7.7 Boron Recycle Evaporator Condensate Monitor

This channel and the discharge piping to the Reactor Makeup Water Storage Tank (RMWST), have been removed from the plant.

11.5.2.7.8 Spent Fuel Pool Demineralizer Monitors

These monitors have been removed from the plant. Sampling of the demineralizer discharge is performed in accordance with Table 11.5-4.

11.5.2.7.9 Spent Fuel Pool Cooling Water Monitor

These monitors have been removed from the plant. Sampling of the spent fuel pool is performed in accordance with Table 11.5-4.

11.5.2.7.10 Turbine Building Sump Monitors

These monitors are provided to detect radioactivity in condensate that has drained into the turbine sumps. Detected radioactivity is indicative of a primary-to-secondary system leak and provides control of potential radioactive effluent discharge. Because this channel monitors the secondary system radioactivity levels, it supplements the main steam line monitors, the steam generator blowdown sample monitor, the condenser off-gas monitor, and the steam generator leak rate monitors. Gamma-sensitive scintillation detectors are provided for the outlet header of each unit's Turbine Building sump pumps. Monitor output indication and alarm annunciation are provided at the Control Room RMS Display/Operator Cabinet. Upon detection of high radiation the monitor trips the valve to stop flow to the Low Volume Waste Pond and directs the discharge to the Waster Water Holdup Tanks (see Section 11.2.2.3).

11.5.2.7.11 Failed-Fuel Monitor

A Geiger-Mueller tube is mounted on the reactor coolant letdown line, after the letdown heat exchanger, to monitor fission-product activity. Detection of increased system activity may be indicative of failed fuel; the monitor initiates an alarm in the Control Room.

11.5.2.7.12 Auxiliary Building to LVW Pond Monitor

Discharges from Auxiliary Building Sumps 3 and 11, the Diesel Generator Sumps, and the CCW Drain Tanks are continuously monitored by a shielded gamma-sensitive (NaI) scintillation detector. If activity concentrations exceed the discharge monitor high radiation alarm setpoint, the discharge is automatically diverted to the Cocurrent Waste System Wastewater Holdup Tanks.

11.5.2.8 Monitor Indication and Alarm

All process and effluent radiation monitors are annunciated, indicated, and have a printout (on demand) of monitor output at the Control Room RMS Display/Operator Cabinet. Containment Air, plant vent stack (PIGs and WRGMs), and control room ventilation intake monitors also have control/display modules in the Control Room. Two distinct, visible alarms are provided for all channels for alert and high alarm. A common audible alarm is provided for alert and high set point trip and for loss-of-instrument signal or background. Certain monitors have local indicators

associated with their channels. Some channels have their indication and annunciation on process system control panels, as listed in Table 11.5-3.

11.5.2.9 Accuracy

The overall accuracy of a radiation monitor system is governed by four major areas: (a) factory calibration and alignment, (b) detector characteristics and environment, (c) microprocessor environment, and (d) field alignment. Factory calibration is performed with standards traceable to the National Institute of Standards and Technology (NIST). Detector energy response and linearity are demonstrated by factory calibration and site calibration procedures subsequent to operations. Detector and microprocessor environmental variations are determined from qualification tests. Overall monitor error is the root-mean-square sum of all system errors and for accident conditions is within a factor of 2 over the entire range.

11.5.2.10 Sensitivity

Threshold sensitivity is defined as twice the background standard deviation.

11.5.2.11 Calibration

The initial calibration of each complete monitoring system was performed by the manufacturer at the factory. These calibrations were primary calibrations performed in accordance with ANSI N13.10 on prototype monitors except for several custom designed monitors. Each process monitor was tested at two or more activity levels with calibration sources traceable to the National Institute of Standards and Technology (NIST). In addition, a linearity test was performed on most monitors to verify that the detection system responds linearly when comparing count rate to activity strength for the operating range of one decade above background to the uppermost decade. The source-to-detector geometry during factory calibration was the same as the sample-to-detector geometry in actual use.

Subsequent calibrations are performed onsite at frequencies specified by the Technical Specifications, Radiological Effluent Controls Program or station procedures. These calibrations are performed using one of the following techniques.

1. Onsite calibrations may be performed using a primary calibration technique similar to the factory calibration described above. This type of calibration is performed using NBS traceable sources or by comparing detector response with the results of laboratory analysis of appropriate grab samples. Onsite primary calibrations may be performed in the Radiation Protection Calibration Facility or in the field.

2. Onsite calibrations may also be performed using portable transfer sources related to an onsite primary calibration. This type of calibration consists of exposing the detector to a transfer calibration source and comparing the detector response to a reference value determined during the onsite primary calibration. Transfer calibrations are performed in the field.

3. A third method of performing onsite calibrations is using transfer sources related to the primary calibration performed at the factory. This type of calibration consists of exposing the detector to a transfer calibration source and comparing the detector response to a

reference value determined during the factory primary calibration. Transfer calibrations are performed in the field.

The calibration technique used for a particular monitor is dependent on several factors including monitor function, complexity of performing a primary calibration onsite, availability of suitable NBS traceable sources, and ALARA concerns. Sources used for calibration, whether primary or transfer, are selected to provide energies representative of the expected average energy of the process stream. The calibration techniques used for each monitor type are specified in approved station procedures.

Each monitor is provided with a check source or check current to check detector response, except for the failed fuel monitor which uses a keep-alive source. For gamma detectors a Cs-137 source is used and for beta detectors a Cl-36 source is used. Requirements for check source tests are discussed in Section 11.5.2.12.

11.5.2.12 Tests and Inspections

11.5.2.12.1 General

The system automatically monitors the operability of every channel by monitoring for conditions such as loss of sample flow, loss of counts from detector and filter torn/out of paper for moving filters. If any of these conditions exist, a channel operate failure alarm is initiated. Additionally, certain critical detectors specified in the plant Technical Specifications 3.4.15.1 or the ODCM are response checked by actuation of the associated check source or current at specified intervals. If the detector fails the check source test, the cause of the failure is investigated. If the investigation determines that the detector or monitor electronics are at fault, the channel is declared inoperable and corrective action to restore the channel to an operable status is initiated.

11.5.2.12.2 Detailed Inspection and Test Procedures

1. The tests and inspections required for the following monitors are listed in the plant Technical Specifications or the ODCM.

   a. Plant Vent Wide Range Gas Monitors (WRGM)

   b. Containment Air PIG Monitors

   c. Liquid Waste Effluent Monitor

   d. Turbine Building Sump Monitors

   e. Service Water Monitors

   f. Control Room Ventilation Intake Monitors

   g. Auxiliary Building to LVW Pond Monitor

   These tests and inspections may consist of source checks, channel checks, digital channel operational tests, and/or channel calibrations. The types and frequencies of

tests required for each of the above monitors is specified in the Technical Specifications or the ODCM.

2. The tests and inspections for all other process monitors are as follows:

   a. Channel trends maintained by the system may be used to evaluate channel behavior.

   b. Channel calibrations are performed at a frequency determined by plant procedure(s), based on performance and function.

   c. During channel calibrations, channel alarm functions and initiation of automatic control functions, if appropriate, are tested by simulation of activity above the setpoint, and by simulating loss of channel power (loss of counts).

11.5.2.13 Maintenance

The channel detector and electronics are serviced and maintained to ensure reliable operations. Manufacturer's recommendations on service and maintenance are available for use. Such maintenance includes cleaning and the replacement or adjustment of any components required after performing a test or calibration check. If any work is performed which could affect the calibration, a recalibration is performed at the completion of the work.

11.5.2.14 Sampling

The radiological sampling system is designed to provide representative liquid and gas samples at controlled temperatures and pressures for laboratory analysis to determine discharge activities and equipment performance and to provide information to assist in operator decision making.

The Process Sampling System (primary) is described in Section 9.3.2. Local effluent sampling and process sampling requirements are discussed in Sections 11.5.3 and 11.5.4, respectively.

Local samples can be obtained from the LWPS and the GWPS at the locations shown in Table 11.5-4. Samples are drawn either batchwise or periodically, as appropriate, for radiological examination. The frequencies of sampling, parameters measured, expected flows, analytical procedures, sensitivities, concentrations, and the basis for selecting each location are delineated in Table 11.5-4. As indicated in the table, the sampling program provides a means to evaluate equipment performance, and in some applications the results determine the further routing of each batch of waste.

Samples are drawn from lines or tanks as grab samples and taken to the radiochemistry laboratory for examination. The radiological sampling monitoring connections are not connected as part of the Process Sampling System (see Section 9.3.2), but the radiological sampling system can furnish radiation information for guidance in process operation.

The guidance of Regulatory Guide 1.21 is followed as described below:

1. The gaseous and liquid monitors are designed to detect the most abundant isotopes of an effluent stream and give indication of activity concentrations.

2. The determination of iodine type and its concentration are provided by an appropriate sampling method and detecting systems

3. Sampling programs will be used in conjunction with a continuous monitoring system to determine total quantity and type of radioisotopes released

4. The proper detecting systems will give the effluent concentration and flow rate

5. Gamma Isotopic analysis from the sampling program will determine isotopic content

6. The meteorological monitoring program provides site environmental data for the dose calculations

The data and parameters obtained from the various subsystems and programs are used to report the necessary information according to the Regulatory Guide 1.21 format.

Sampling techniques are discussed further in Section 9.3.2.

11.5.3 EFFLUENT MONITORING AND SAMPLING

The requirement of GDC 64 for the monitoring of effluent discharge paths is implemented by providing continuous radiation detection and/or periodic sampling for all liquid and gaseous effluent paths from which detectable quantities of radioactivity can be released from the plant during normal operation, including anticipated operational occurrences and accidents. The criteria for locating radiation monitors and sampling points are discussed in Sections 11.5.2.2 and 11.5.2.14, respectively. Effluent sampling of all potentially radioactive liquid and gaseous effluent paths is conducted on a regular basis to verify the adequacy of effluent processing to meet the discharge limits. The effluent sampling program provides the information for the effluent measuring and reporting programs required by 10 CFR Part 50.36a in annual reports to the NRC. In general, the effluent sampling program requires that stored wastes be sampled and analyzed prior to release to the environment, and that potentially radioactive, continuously released effluents be periodically sampled and analyzed. Sampling and analysis of each effluent release pathway is performed in accordance with the requirements established in the Radiological Effluent Controls Program required by the Technical Specifications.

Table 11.5-4 summarizes the sampling and analysis requirements for each release pathway. The requirements included in this summary are considered normal. Requirements for increased sampling frequencies and allowances for exemptions to the sampling requirements are included in the Radiological Effluent Controls sampling program required by the Technical Specifications.

The following paragraphs summarize the categories of effluent release types, list the specific pathways associated with each category, and provide a general discussion of sampling and analysis requirements.

11.5.3.1 Radioactive Liquid Effluents Released on a Batch Basis to the Circulating Water Discharge

Release pathways in this category include discharges from the Waste Monitor Tanks, the Laundry Holdup and Monitor Tanks, Waste Water Holdup Tanks and the Plant Effluent Holdup and Monitor Tanks. Gamma isotopic measurements are made on each batch of effluent released

and kept as a record together with the volume of the batch, the average dilution water flow used during discharge, and the date and time of release. Additionally, samples from each batch are composited and periodically analyzed for difficult to measure isotopes (i.e., alpha emitters and beta emitters). Composite samples are samples in which the quantity of liquid added to the composite from each batch released is proportional to the quantity of liquid in that batch. The results of sample analyses are used for effluent accountability and to ensure that release limits are not exceeded.

The Waste Water Holdup Tanks (WWHTs) discussed above are discharged to the Circulating Water Discharge when activity levels exceed limits for discharge to the LVW Pond specified in the Radiological Effluent Controls Program. If secondary wastes are transferred to the WWHTs when such diversion is not required by the Radiological Effluent Controls Program, the tanks may be discharged to the Low Volume Waste Pond. Discharges via this pathway are only allowed when the sample analysis shows that radioactivity concentrations do not exceed specified levels.

11.5.3.2 Potentially Radioactive Liquid Effluents Released Directly to the Environment

1. Continuous Releases

   The release pathways in this category are the discharges from the Turbine Building Sumps, Auxiliary Building Sumps 3 and 11, Diesel Generator Sumps and the Component Cooling Water Drain Tanks to the Low Volume Waste Pond. These waste streams are required to be sampled if detectable quantities of radioactivity are present. They are continuously monitored by process radiation monitors. If a monitor is out of service, then periodic sampling maybe performed. If radioactivity is present above specified levels, the waste stream is diverted to the Waste Water Holdup Tanks where it is isolated, sampled, and released as discussed in Section 11.5.3.1, above. Additionally, Low Volume Waste Pond discharges to Squaw Creek Reservoir are sampled for final accountability of radioactive materials that have been released to the pond.

2. Batch Releases

   The release pathway in this category is the discharge from the Condensate Backwash Recovery Tanks to the Low Volume Waste Pond. These tanks are sampled in accordance with the Radiological Effluent Controls Program. If radioactivity is present above specified levels, the tank discharge is diverted to the Waste Water Holdup Tanks where it is isolated, sampled, and released as discussed in Section 11.5.3.1, above.

11.5.3.3 Radioactive Gaseous Effluents Released on Batch Basis to the Environment

Release pathways in this category include intentional discharges from the containment purge exhaust and the Waste Gas Decay Tanks via the plant vent stacks. These confinements are sampled and analyzed prior to release. Additionally, the releases are continuously monitored by the plant vent monitors. The Plant Vent Stack particulate and iodine filters are changed and analyzed at least weekly to determine the quantities of radionuclides with half-lives greater than 8 days and radioiodine released, respectively. The results of sample analyses are used for effluent accountability and to ensure that release limits are not exceeded.

11.5.3.4 Radioactive Gaseous Effluents Released Continuously to the Environment

The release pathway in this category is the routine continuous ventilation system discharge via the plant vent monitors. Potentially radioactive gases are continuously discharged from the Fuel Building, Safeguards Building, and Auxiliary Building ventilation exhaust systems, and the Condenser Off-Gas System via the plant vent stacks. A representative sample from the plant vent stacks is collected and analyzed for noble gas isotopes at least monthly. A continuous sample from the plant vent is drawn through particulate and iodine filters. These filters are changed and analyzed at least weekly. The particulate filters are also composited and periodically analyzed for more difficult to measure isotopes (i.e., alpha emitters and pure beta emitters). The sampling frequencies are increased during refueling outages and after process changes or other occurrences that could alter the mixture of radionuclides. The results of sample analyses are used for effluent accountability and to ensure that release limits are not exceeded.

11.5.3.5 Analytical Procedures

Samples of process and effluent gases and liquids are analyzed in the laboratory (onsite or offsite) by the following techniques:

1. Gross beta counting

2. Gross alpha counting

3. Gamma spectrometry

4. Liquid scintillation counting

5. Radiochemical separations (by in-house or outside laboratory)

Instrumentation, which is available in the laboratory for the measurement of radioactivity, includes the following:

1. Proportional counter

2. Liquid scintillation counter

3. Gamma Spectrometer

    a. HP(Ge) detector, coupled with

    b. Multichannel analyzer with interfaced computer

Gross alpha analyses of liquid effluent samples and liquid process samples are performed with the proportional counter. These samples are evaporated to dryness on planchets prior to counting. Sample volume, counting geometry, and counting time are chosen to achieve the required measurement sensitivities. Correction factors are applied for sample decay, counter-resolving time, and other parameters as necessary to assure accuracy.

Gross alpha analyses of air particulate samples are performed by direct counting of the filters with the proportional counter.

Gross beta analyses of air particulate or liquid samples may be performed in the laboratory using the capabilities of the proportional counter or by employing counting techniques using beta scintillation detection.

Gamma spectrometry is used extensively for isotopic analyses of gaseous, air particulate, and liquid samples. High-resolution, high-purity Germanium detectors are available for this purpose. All detectors are calibrated against gamma energy for a variety of sample detector geometries.

Liquid samples for tritium analysis are normally purified as necessary prior to analysis by distilling the samples. The liquid scintillation counter is used to count the samples.

Radiochemical separations are performed by an approved outside laboratory or in-house for the purpose of the routine analysis of Sr-89 and Sr-90.

11.5.4 PROCESS MONITORING AND SAMPLING

The requirement of GDC 60 is to provide for control of the releases of radioactivity to the environment. The requirement of GDC 60, with respect to suitably controlling releases from gaseous and liquid effluent discharge paths, is implemented by providing monitors (as described in Section 11.5.2) that detect excessive radiation levels.

If radiation levels in the liquid discharge lines exceed predetermined monitor set points, the monitor trips and initiates closure of the discharge isolation valve to terminate release. If radiation levels in the primary plant ventilation system exceed predetermined monitor setpoints, the monitor trips and initiates closure of the gaseous waste processing system release valve.

GDC 63 requires the provision of instrumentation to detect excessive radiation levels in fuel storage and radioactive waste processing systems and associated handling areas. This requirement is implemented by the use of the ARMS, the PRMS, the portable monitoring and sampling system, and the process sampling program. The process radiation monitoring instrumentation detects and monitors gamma radiation contained in fluids of the Liquid Waste Processing System, and the Gaseous Waste Processing System. Increased fluid activity concentrations in these systems are promptly sensed by the monitors because the monitors are shielded for background levels of radioactivity. The small fluid leakage from the Waste Processing Systems (WPS) provides low airborne radioactivity concentrations. When there are large increases of WPS fluid activities, the leaking fluid provides higher airborne activities which are detected by the ventilation and area monitoring systems.

Process streams of the various subsystems of the Liquid Waste Processing System have monitors that are augmented by a sampling system, and this combination provides radiation monitoring to determine changing levels of system radioactivity and equipment performance.

Proper sampler system design for the medium monitored is provided to assure representative samples from radioactive process streams and tanks.

Provisions for sampling from ducts and stacks are in agreement with ANSI N13.1.

TABLE 11.5-1
PROCESS RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 1 of 5)

Detector Nos.[a]									
Unit 1	Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/cm3)	Bases for Alarm Set Points
Auxiliary Building									
XRE-5567A XRE-5567B		Beta scintillator	Plant vent effluent -- noble gases (off-line)	El. 873 feet 6 in. Fig. 12.3-17	Kr-85, Xe-135 Xe-133[b]	Air	Gross beta	1E-06 to 1E-02	Note 2
XRE 5570A XRE 5570B		Beta scintillator CdTe(Cl)	Plant vent effluent -- noble gases (off-line)	El. 873 feet 6 inches Figures: 12.3-8 12.3-9, 12.3-23.6, and 12.3-23.7 Flow Diag. Fig. 9.4-9	Kr-85, Xe-135 Xe-133[b], Kr-88, Xe-138	Air	Gross beta 1E+05	1E-06 to 1E+05	Note 2
XRE 5701	-	Beta scintillator	Auxiliary Building ventilation air exhaust - noble gases (in-line)	Vent duct El. 873 feet 6 in. Fig. 12.3-17 Flow Diag. Fig. 9.4-2	Kr-85, Xe-135 Xe-133[b]	Air	Gross beta	1E-04 to 1E 00	Note 2, 4
1RE 5637	2RE 5637	Beta scintillator	Main steam and feedwater area ventilation air exhaust noble gases (in-line)	Vent duct El. 872 feet 6 in. Figures: 12.3-8 and 12.3-23.6 Flow Diag. Fig. 9.4-4	Kr-85, Xe-135 Xe-133[b]	Air	Gross beta	1E-04 to 1E 00	Note 5
XRE 5250	-	Gamma scintillator	Aux. Bldg. Waste Gas Adjacent to line	GWPS El. 862 feet 6 in. Fig. 12.3-16 Flow Diag. Fig. 11.3-1	Kr-85, Xe-135 Xe-133[b]	Gas	Gross gamma	1E-01 to 1E+04	Note 1
1RE 4269 1RE 4270	2RE 4269 2RE 4270	Gamma scintillator	Aux. Bldg. Service Water (off-line)	El. 790 feet 6 in. Fig. 12.3-12 Flow Diag. Fig. 9.2-1	I-131, I-133 Cs-134, Cs-137, Co-58, Co-60[b]	Water	Gross gamma 5E-02	1E-05 to 5E-02	Note 2
1RE 4509 1RE 4510 1RE 4511	2RE 4509 2RE 4510 2RE 4511	Gamma scintillator	Component cooling water (off-line)	El. 810 feet 6 in. Fig. 12.3-13 Flow Diag. Fig. 9.2-3	1-131, I-133 Cs-134, Cs-137, Co-58, Co-60[b]	Water	Gross gamma	1E-05 to 5E-02	Note 7

TABLE 11.5-1
PROCESS RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 2 of 5)

Detector Nos.[a]									
Unit 1	Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/cm3)	Bases for Alarm Set Points
XRE 5253		Gamma scintillator	Effluent Liquid Waste (off-line)	El. 790 feet 6 in. Fig. 12.3-12 Flow Diag. Fig. 11.2-5	I-131, I-133, Cs-134, Cs-137, Co-58, Co-60[b]	Water	Gross gamma	1E-05 to 5E-02	Note 2
XRE 5251A		Gamma scintillator	Auxiliary Bldg. to LVW Pond Liquid Waste Effluent (adjacent-to-line)	El. 778 feet Fig. 12.3-21.2 Flow Diag. Fig. 9.3-8	I-131,I-133 Cs-134, CS-137 Co-58, Co-60[b]	Water	Gross gamma	1E-05 to 5E-02	Note 2
1RE 5698	2RE 5698	Beta scintillator	Safeguards building ventilation air (in-line)	Vent duct El. 873 feet. 6 in. Fig. 12.3-17 Flow Diag. Fig. 9.4-2	Xe-133[b], Xe-135 Kr-85	Air	Gross beta	1E-04 to 1E-00	Note 4
XRE 5700	-	Beta scintillator	Fuel building ventilation air (in-line)	Vent duct. El. 886 feet Fig. 12.3-17 Flow Diag. Fig. 9.4-2	Xe-133[b], Xe-135 Kr-85	Air	Gross beta	1E-04 to 1E-00	Note 4
XRE 5702	-	Beta scintillator	HVAC room ventilation air (in-line)	Vent duct El. 873 feet 6 in. Fig. 12.3-17 Flow Diag. Fig. 9.4-2	Xe-133[b], Xe-135 Kr-85	Air	Gross beta	1E-04 to 1E-00	Note 4
Safeguards Building									
1RE 4200	2RE 4200	Gamma scintillator	Steam generator blowdown sample (off-line)	El. 810 feet 6 in. Figures 12.3-6 and 12.3-23.4 Flow Diag. Fig. 9.3-4	I-131, I-133, Cs-134, Cs-137 Co-58, Co-60[b]	Water	Gross gamma	1E-05 to 5E-02	Note 3
1RE 5502	2RE 5502	Beta scintillator	Containment air-particulate (off-line)	El. 831 feet 6 in. Figures 12.3-7 and 12.3-23.5 Flow Diag. Fig. 9.4-6	Cs-137[b], Rb-88, I-133	Air	Gross beta	5E-11 to 5E-07	Note 7

TABLE 11.5-1
PROCESS RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 3 of 5)

Detector Nos.[a]									
Unit 1	Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/cm3)	Bases for Alarm Set Points
1RE 5566	2RE 5566	Gamma scintillator	Containment air iodine (off-line)	El. 831 feet 6 in. Figures 12.3-7 and 12.3-23.5 Flow Diag. Fig. 9.4-6	I-131[b], I-133	Air	Isotopic - I-131	Note 6	Note 4
1RE 5503	2RE 5503	Beta scintillator	Containment air noble gas (off-line)	El. 831 feet 6 in. Figures 12.3-7 and 12.3-23.5 Flow Diag. Fig. 9.4-6	Xe-133[b], Kr-85 Xe-135	Air	Gross beta	1E-06 to 1E-02	Note 2
1RE 406	2RE 406	Geiger-Mueller tube	Reactor coolant letdown line liquid / failed fuel (off-line)	El. 831 feet 6 in. Figures 12.3-7 and 12.3-23.5 Flow Diag. Fig. 9.3-10	Co-60[b], Co-58 Cs-134, Cs-137	Water	Gross gamma	1E-00 to 1E+05	Note 1
1RE 2325	2RE 2325	Geiger-Mueller tube	Main steam noble gas (on-line)	El. 873 feet 6 in. Figures 12.3-9 and 12.3-23.7 Flow Diag. Fig. 10.3-1	Xe-133[b], Kr-85 Xe-135	Steam	Gross gamma	1E-01 to 1E+03	Note 3
1RE 2326	2RE 2326	Geiger-Mueller tube	Main steam noble gas (on-line)	El.873 feet 6 in. Figures 12.3-9 and 12.3-23.7 Flow Diag. Fig. 10.3-1	Xe-133[b], Kr-85 Xe-135	Steam	Gross gamma	1E-01 to 1E+03	Note 3
1RE 2327	2RE 2327	Geiger-Mueller tube	Main steam noble gas (on-line)	El. 873 feet 6 inches Figures 12.3-9 and 12.3-23.7 Flow Diag. Fig. 10.3-1	Xe-133[b], Kr-85 Xe-135	Steam	Gross gamma	1E-01 to 1E+03	Note 3
1RE 2328	2RE 2328	Geiger-Mueller tube	Main steam noble gas (on-line)	El. 873 feet 6 in. Figures 12.3-9 and 12.3-23.7 Flow Diag. Fig. 10.3-1	Xe-133[b], Kr-85 Xe-135	Steam	Gross gamma	1E-01 to 1E+03	Note 3

TABLE 11.5-1
PROCESS RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 4 of 5)

Detector Nos.[a]									
Unit 1	Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/cm3)	Bases for Alarm Set Points
IRE 2325A	2RE2325A	Gamma scintillator	N-16 (on-line)	El. 873 feet 6 in. Figure 12.3-9 and 12.3-23.7 Flow Diag. Fig. 10.3-1	N-16	Steam	Gamma	1E-03 to 1E+03 Note 9	Note 3
IRE 2326A	2RE2326A	Gamma scintillator	N-16 (on-line)	El. 873 feet 6 in. Figure 12.3-9 and 12.3-23.7 Flow Diag. Fig. 10.3-1	N-16	Steam	Gamma	1E-03 to 1E+03 Note 9	Note 3
IRE 2327A	2RE2327A	Gamma scintillator	N-16 (on-line)	El. 873 feet 6 in. Figure 12.3-9 and 12.3-23.7 Flow Diag. Fig. 10.3-1	N-16	Steam	Gamma	1E-03 to 1E+03 Note 9	Note 3
IRE 2328A	2RE2328A	Gamma scintillator	N-16 (on-line)	El. 873 feet 6 in. Figure 12.3-9 and 12.3-23.7 Flow Diag. Fig. 10.3-1	N-16	Steam	Gamma	1E-03 to 1E+03 Note 9	Note 3
Electrical and Control Building									
XRE 5895A XRE 5896A XRE 5895B XRE 5896B		Beta scintillator	Control Room ventilation intake (off-line)	El. 854 feet 4 in. El. 852 feet 6 in. Fig. 12.3-16 Flow Diag. Fig. 9.4-1	Xe-133[b], Xe-135 Kr-85, I-131	Air	Gross beta	1E-06 to 1E-02	Note 4
Turbine Building									
1RE 2959	2RE 2959	Beta scintillator	Condenser off-gas (off-line)	El. 803 feet Figures 12.3-23 and 12.3-23.1 Flow Diag. Fig 10.4-3	Kr-85, Xe-133[b] Xe-135	Gas	Gross beta	1E-05 to 1E-01	Note 3

TABLE 11.5-1
PROCESS RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 5 of 5)

Detector Nos.[a]									
Unit 1	Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/cm3)	Bases for Alarm Set Points
1RE 5100	2RE 5100	Gamma scintillator	Turbine Building drains liquid (Adjacent-to-line)	El. 775 feet 3 in. Figures 12.3-21.1 and 12.3-22.1 Flow Diag. Fig 9.3-8	Co-60[b], Cs-134, Co-58, Cs-137	Water	Gross gamma	1E-05 to 5E-02	Note 2

a) Detector numbers preceded by an “X” (i.e., XRE 5701) are common to both units.

b) Reference nuclide

NOTES:

1. Alarm set points are based on process system requirements for normal operation.
2. Alarm set points are based on effluent requirements for normal operation.
3. Alarm set points are based on Steam Generator Tube Rupture (SGTR) detection.
4. Alarm set points are based on airborne radioactivity considerations for personnel protection.
5. Alarm set points are based on indication of containment integrity for accident conditions.
6. Since these iodine detectors monitor the accumulated activity of iodine in the iodine adsorber cartridge, the specified instrument range (uCi/Cm3) and MDC (uCi/Cm3) are dependent on the frequency of the cartridge replacement, flow rate, sensitivity, and detector count rate. The activity concentration may be calculated as follows:

$$\text{Activity } (\mu Ci/cm^3) = \frac{\text{Count Rate (counts/min)}}{\text{Time since cartridge replacement (min)} \times \text{Sample flow rate } (cm^3/min) \times \text{Fractional cartridge efficiency} \times \text{Sensitivity (counts per min/uCi)}}$$

The sensitivity of the detector is > 4E+4 cpm/uCi.

7. Alarm setpoints are based upon RCS pressure boundary leakage detection.
8. GM tube (RD-10) detector activity between 1E-03 and 1E-01 is outside the 2 sigma, 95% confidence level delineated by ANSI 13.10.
9. This indicates sensitivity range of detector. The readout is in gallons per day.

TABLE 11.5-2
PROCESS RADIATION MONITORING SYSTEM ENVIRONMENTAL DESIGN CONDITIONS

Item	Temperature Range (F)		Relative Humidity (%)		Pressure (in. psig)	
	Normal	Accident	Normal	Accident	Normal	Accident
Airborne, Steam and Gaseous Channels						
a. Control Room electronics, printers and CRT displays	70 to 80 (air)	80 (air)	50	50	0.1	0.1
b. Local electronics and sampling equipment (pumps, motors, and so forth)	60 to 104 (air)	122 (air)	50 to 100	100	-0.1	Atmospheric
c. Detectors (except below)	40 to 122 (air)	122 (air)	50 to 100	100	-0.1	Atmospheric
d. Detector XRE 5250	40 to 180 (air)	180 (air)	50 to 100	100	-0.1	Atmospheric
e. Detectors RE 2325, 2326, 2327, and 2328	40 to 160 (air)	165 (air)	70	95		Atmospheric
f. Detectors RE 2325A, 2326A, 2327A, and 2328A	40 to 120 (air)	140 (air)	70	N/A		Atmospheric
Liquid Channels						
a. Control Room electronics, printer and CRT displays	70 to 80 (air)	80 (air)	50	50	0.1	0.1
b. Local electronics and sampling equipment (pumps, motors, and so forth)	60 to 104 (air)	122 (air)	50 to 100	100	-0.1	Atmospheric
c. Detector for Monitor RE 4200	40 to 122 (Liquid)	122 (Liquid)	N/A	N/A	--	--
d. Deleted						
e. Deleted						
f. Detectors for Monitors RE 4269, RE 4270, RE 4509 RE 4510, RE 4511	32 to 104 (Liquid)	122 (Liquid)	N/A	N/A	--	--
g. Detector for Monitor RE-5251A, RE-5100	40 to 180 (Liquid)	180 (Liquid)	N/A	N/A	--	--
h. Detector for Monitor RE-5253	32 to 122 (Liquid)	122 (Liquid)	N/A	N/A	--	--
i. Detector for Monitor RE-406	40-140 (Liquid)	140 (Liquid)	N/A	N/A	--	--

TABLE 11.5-3
PROCESS RADIATION MONITORING SYSTEM CHARACTERISTICS

(Sheet 1 of 5)

Monitor	Channel Nos.		Indicator and Alarm Types		
	Unit 1	Unit 2	Local (Monitor System)	Remonte (Control Room)	Monitor Automatic Control Actions
Plant vent stack (noble gas)	XRE 5567A	XRE 5567B	None	Console (CRT, printer) with audible and Visible alarm (alert, high, and failure) and rack mounted operator panel with visible alarm and operator console CRT with visible alarm (alert, high and failure)	None
Plant vent stack (noble gas)	XRE 5570A	XRE 5570B	None	"	High radiation signal (Note 1) initiatives closure of valve HCV-014 in the GWPS.
Containment (particulate)	1RE 5502	2RE 5502	None	Console (CRT, printer) with audible and visible alarm (alert, high, and failure) and seismic panel with visible alarm and operator console CRT with visible alarm (alert, high and failure)	High radiation signal initiates Containment ventilation isolation.
Containment (iodine)	1RE 5566	2RE 5566	None	"	None

TABLE 11.5-3
PROCESS RADIATION MONITORING SYSTEM CHARACTERISTICS

(Sheet 2 of 5)

	Channel Nos.		Indicator and Alarm Types		
Monitor	Unit 1	Unit 2	Local (Monitor System)	Remonte (Control Room)	Monitor Automatic Control Actions
Containment (noble gas)	1RE 5503	2RE 5503	None	"	High radiation signal initiates Containment ventilation isolation.
Control Room Ventilation Intake	XRE 5895A XRE 5896A XRE 5895B XRE 5896B		None	Display/Operator Cabinet (LCD Display, printer) with audible and visible alarm (alert, high, and failure) and seismic/ Class 1E panel with visible alarm and operator console CRT with visible alarm (alert, high and failure)	High radiation signal initiates Control Room ventilation emergency recirculation through filters
Condenser off-gas	1RE 2959	2RE 2959	None	Display/Operator Cabinet (LCD Display, printer) with audible and visible alarm (alert, high, and failure) and operator console CRT with visible alarm (alert, high and failure)	None
Waste Gas	XRE 5250		GWPS panel, (ratemeter and alarm)	"	"

TABLE 11.5-3
PROCESS RADIATION MONITORING SYSTEM CHARACTERISTICS

(Sheet 3 of 5)

Monitor	Channel Nos.		Indicator and Alarm Types		Monitor Automatic Control Actions
	Unit 1	Unit 2	Local (Monitor System)	Remonte (Control Room)	
Steam generator blowdown sample	1RE 4200	2RE 4200	Steam Generator Blowdown Sample panel (alert alarm)	Display/Operator Cabinet (LCD Display, printer) with audible and visible alarm (alert, high and failure) and operator console CRT with visible alarm (alert, high and failure)	High radiation signal closes isolation valves in the blowdown and sample lines.
Auxiliary Building to LVW Pond	XRE 5251A		CPX-ECPRLV-01 (Auxiliary Bldg. to pond radiation detector control panel)	"	High radiation signal closes valve to LVW pond and diverts discharge to Cocurrent Wastewater Holdup Tanks
Waste Liquid	XRE 5253		LWPS panel (ratemeter and alert alarm)	"	High radiation signal closes discharge valve to circulating water discharge (Note 1)
Service Water	1RE 4269	2RE 4269	None	"	None
Service Water	1RE 4270	2RE 4270	None	Display/Operator Cabinet (LCD Display, printer) with audible and visible alarm (alert, high, and failure) and operator console CRT with visible alarm (alert, high and failure)	None

TABLE 11.5-3
PROCESS RADIATION MONITORING SYSTEM CHARACTERISTICS

(Sheet 4 of 5)

Monitor	Channel Nos.		Indicator and Alarm Types		
	Unit 1	Unit 2	Local (Monitor System)	Remonte (Control Room)	Monitor Automatic Control Actions
Component cooling water	1RE 4509	2RE 4509	None	"	"
Component cooling water	1RE 4510	2RE 4510	None	"	"
Component cooling water	1RE 4511	2RE 4511	None	"	"
Turbine building drains	1RE 5100	2RE 5100	Turbine Bldg. Sump Pump Control Panel (CP1-EIPRLV-25)	Display/Operator Cabinet (LCD Display, printer) with audible and visible alarm (alert, high, and failure) and operator console CRT with visible and alarm (alert, high and failure)	High radiation signal closes discharge valve to low volume waste pond directs discharge to a wastewater holdup tank. (Note 1)
Auxiliary Building vent duct	XRE 5701		Gaseous Waste Processing System (GWPS) Panel (ratemeter and alert alarm)	"	High radiation signal closes valve HCV-014 in GWPS (Note 1).
HVAC room vent duct	XRE 5702		None	"	None
Fuel Building vent duct	XRE 5700		None	"	None

TABLE 11.5-3
PROCESS RADIATION MONITORING SYSTEM CHARACTERISTICS

(Sheet 5 of 5)

Monitor	Channel Nos.		Indicator and Alarm Types		
	Unit 1	Unit 2	Local (Monitor System)	Remonte (Control Room)	Monitor Automatic Control Actions
Safeguards Building vent	1RE 5698	2RE 5698	None	"	None
Main Steam and FW Area	1RE 5637	2RE 5637	None	"	None
Failed fuel	1RE 406	2RE 406	None	"	None
Main Steam Line	1RE 2325 1RE 2326 1RE 2327 1RE 2328	2RE 2325 2RE 2326 2RE 2327 2RE 2328	None	"	None
Steam Generator Leak Rate Monitor	1RE 2325A 1RE 2326A 1RE 2327A 1RE 2328A	2RE 2325A 2RE 2326A 2RE 2327A 2RE 2328A	None	"	None

Note:

1. In addition to the high alarm signal, circuit failure, loss of counts, loss of flow, or channel out of service will initiate the control function.

TABLE 11.5-4
PROCESS AND EFFLUENT RADIOLOGICAL SAMPLING SYSTEM CHARACTERISTICS

(Sheet 1 of 6)

SAMPLING LOCATION	SAMPLING FREQUENCY	SAMPLE TYPE	EXPECTED SAMPLE AMOUNT (ml)	EXPECTED SAMPLE CONCENTRATIONS	ANALYSIS PERFORMED	REQUIRED LOWER LIMIT OF DETECTION (micro Ci/ml)	SAMPLE PROCESS
Chemical and Volume Control System (CVCS)							
Recycle holdup tanks	As Required	Liquid Grab Sample	60	Primary Coolant	Gamma Spec.	N.A.	To determine purification requirements of waste and evaluate performance of recycle evaporator
Upstream of recycle evaporator feed demineralizers	As Required	Liquid Grab Sample	60	Primary Coolant	Gamma Spec.	N.A.	To evaluate performance of recycle evaporator feed demineralizer
Downstream of recycle evaporator feed demineralizer	As Required	Liquid Grab Sample	60	Primary Coolant	Gamma Spec.	N.A.	To evaluate performance of evaporator feed demineralizer
Recycle evaporator distillate	As Required	Liquid Grab Sample	60	Primary Coolant	Gamma Spec.	N.A.	To evaluate performance of recycle evaporator
Downstream of recycle evaporator condensate demineralizer	As Required	Liquid Grab Sample	60	Primary Coolant	Gamma Spec.	N.A.	To evaluate performance of recycle evaporator condensate demineralizer
Recycle Evaporator concentrates	As Required	Liquid Grab Sample	60	Primary Coolant	Gamma Spec	N.A.	To determine whether to reprocess, recycle or solidify
Refueling water tank	As Required	Liquid Grab Sample	60	Negligible	Gamma Spec.	N.A.	General monitoring

TABLE 11.5-4
PROCESS AND EFFLUENT RADIOLOGICAL SAMPLING SYSTEM CHARACTERISTICS

(Sheet 2 of 6)

SAMPLING LOCATION	SAMPLING FREQUENCY	SAMPLE TYPE	EXPECTED SAMPLE AMOUNT (ml)	EXPECTED SAMPLE CONCENTRATIONS	ANALYSIS PERFORMED	REQUIRED LOWER LIMIT OF DETECTION (micro Ci/ml)	SAMPLE PROCESS
CCWS							
Downstream of the CCW pumps	As Required	Liquid Grab Sample	500	Negligible	Gamma Spec.	N.A.	To check on CCWS radiation monitors
SWS							
Downstream of the CCW heat exchangers	As Required	Liquid Grab Sample	500	Negligible	Gamma Spec.	N.A.	To monitor any in-leakage from the CCWS
Recyclable Waste Treatment System							
Waste holdup tank	As Required	Liquid Grab Sample	60	Table 11.2-2	Gamma Spec.	N.A.	To determine processing requirements, evaluate performance of waste evaporator
Waste evaporator distillate	As Required	Liquid Grab Sample	60	Will Vary	Gamma Spec.	N.A.	To evaluate performance of waste evaporator
Downstream of waste evaporator distillate demineralizer	As Required	Liquid Grab Sample	60	Will Vary	Gamma Spec.	N.A.	To evaluate performance of waste evaporator condensate demineralizer
Waste condensate tank	As Required	Liquid Grab Sample	60	Will Vary	Gamma Spec.	N.A.	To determine disposition of processed condensate
Waste evaporator concentrates	As Required	Liquid Grab Sample	60	Will Vary	Gamma Spec.	N.A.	To determine whether to recycle, return to BRS or to solid waste management system

TABLE 11.5-4
PROCESS AND EFFLUENT RADIOLOGICAL SAMPLING SYSTEM CHARACTERISTICS

(Sheet 3 of 6)

SAMPLING LOCATION	SAMPLING FREQUENCY	SAMPLE TYPE	EXPECTED SAMPLE AMOUNT (ml)	EXPECTED SAMPLE CONCENTRATIONS	ANALYSIS PERFORMED	REQUIRED LOWER LIMIT OF DETECTION (micro Ci/ml)	SAMPLE PROCESS
Non-Reactor Grade Waste Treatment System							
Laundry and hot shower tank	As Required	Liquid Grab Sample	1000	Negligible	Gamma Spec.	N.A.	To determine whether to process or discharge
Upstream of waste monitor tank demineralizer	As Required	Liquid Grab Sample	1000	Will Vary	Gamma Spec.	N.A.	To evaluate performance of demineralizer
Downstream of waste waste monitor tank demineralizer	As Required	Liquid Grab Sample	1000	Will Vary	Gamma Spec.	N.A.	To evaluate performance of demineralizer
Primary Water Makeup System							
Reactor Makeup Water Storage tank	As Required	Liquid Grab Sample	1000	Will Vary	Gamma Spec.	N.A.	Evaluation of systems for recycling waste water
Spent Fuel Pool Cooling and Demineralizer System							
Spent Fuel Pool	Weekly	Liquid Grab Sample	60	Will Vary	Gamma Spec.	N.A.	To determine purification requirements; evaluate leakage from spent fuel
Demineralizer Effluent	Weekly	Liquid Grab Sample	60	Will Vary	Gamma Spec.	N.A.	To determine purification performance
Condensate and Feedwater System							

TABLE 11.5-4
PROCESS AND EFFLUENT RADIOLOGICAL SAMPLING SYSTEM CHARACTERISTICS

(Sheet 4 of 6)

SAMPLING LOCATION	SAMPLING FREQUENCY	SAMPLE TYPE	EXPECTED SAMPLE AMOUNT (ml)	EXPECTED SAMPLE CONCENTRATIONS	ANALYSIS PERFORMED	REQUIRED LOWER LIMIT OF DETECTION (micro Ci/ml)	SAMPLE PROCESS
Steam generator blowdown or other secondary sample point	Per Tech Spec	Liquid Grab Sample	1000	Negligible	Dose Equivalent I-131	N.A.	Primary to Secondary leak determination
Pressurizer relief tank vapor space and/ or Pressurize Vapor Space	As Required	Gas Grab Sample	30	Concentrations of individual isotopes relative to one another is proportional to entrained noble gases in primary coolant.	Gamma Spec.	N.A.	To evaluate fuel cladding (PRT) integrity, PRT leakage, and necessity for PRT purging
Auxiliary Steam System							
Condensate	As Required	Liquid Grab Sample	1000	Will Vary	Gamma Spec.	N.A.	To monitor for radwaste evaporator tube leakage
Reactor Coolant Drain Tank	As Required	Gas Grab Sample	30	Concentrations of individual isotopes relative to one another is proportional to entrained noble gases in primary coolant.	Gamma Spec.	N.A.	To determine whether to recycle or process
Recycle Holdup Tanks	As Required	Gas Grab Sample	30	Concentration of individual isotopes relative to one another is proportional to entrained noble gases in primary coolant.	Gamma Spec.	N.A.	To evaluate tank leakage; determine whether to purge to GWPS
Radioactive Effluent Sampling							

TABLE 11.5-4
PROCESS AND EFFLUENT RADIOLOGICAL SAMPLING SYSTEM CHARACTERISTICS

(Sheet 5 of 6)

SAMPLING LOCATION	SAMPLING FREQUENCY	SAMPLE TYPE	EXPECTED SAMPLE AMOUNT (ml)	EXPECTED SAMPLE CONCENTRATIONS	ANALYSIS PERFORMED	REQUIRED LOWER LIMIT OF DETECTION (micro Ci/ml)	SAMPLE PROCESS
Waste gas decay tanks	Per ODCM	Gas Grab Sample	30	Table 11.3-4 Gamma	Principal emitters	1.0E-04	Effluent monitoring
Containment atmosphere	Per ODCM	Gas and H-3 Grab Sample	30 (gas) 100 liters (H-3)	Table 12.2-26	Principal gamma emitters and tritium	1.0E-04 (gas) 1.0E-06 (H-3)	Effluent monitoring
Plant Vent collection header	Per ODCM	Gas, H-3, Particulate, charcoal or silver zeolite grab sample	1000(gas), 100 liters (H-3), The remaining samples are dependent upon operation	Concentrations will vary depending upon effluent releases occurring, other gaseous process considerations and airborne contaminants.	Principal gamma emitters, H-3 I-131, Sr-89 Sr-90, Gross Alpha Noble gas	1.0E-4 (gas) 1.0E-12 (I-131) 1.0E-6 (H-3) 1.0E-11 (Particulate, Gross alpha, Sr-89, Sr-90) 1.0E-6 (Noble gas)	Effluent monitoring
Liquid Process Systems							
Radioactive waste processing system (Waste Monitor Tanks & Laundry Holdup & Monitor Tanks)	Per ODCM	Liquid Grab Sample	500	Will vary Sr-89, Sr-90 Fe-55	Principal gamma emitters, I-131, Dissolved and entrained gasses, H-3, Gross alpha (Sr-89, Sr-90)	5.0E-7 (Principle Gamma Emitters) 1.0E-6 (I-131, Fe-55) 1. 0E-7 (Gross Alpha) 5.0E-8 1.0E-5 (dissolved and entrained noble gases, H-3)	Effluent monitoring or to determine whether to reprocess
Condensate polisher backwash recovery tanks & component cooling water drain tanks and turbine building sumps	Per ODCM	Liquid Grab Sample	1000	Negligible but can vary	Principal Gamma emitters I-131, H-3	1.OE-06 (I-131) 5.OE-07 (Principle Gamma Emitters) 1.OE-05 (H-3)	Effluent monitoring

TABLE 11.5-4
PROCESS AND EFFLUENT RADIOLOGICAL SAMPLING SYSTEM CHARACTERISTICS

(Sheet 6 of 6)

SAMPLING LOCATION	SAMPLING FREQUENCY	SAMPLE TYPE	EXPECTED SAMPLE AMOUNT (ml)	EXPECTED SAMPLE CONCENTRATIONS	ANALYSIS PERFORMED	REQUIRED LOWER LIMIT OF DETECTION (micro Ci/ml)	SAMPLE PROCESS
Low volume Waste Pond discharges	Per ODCM	Composite Sample over discharge period	1000	Negligible but can vary	Principal gamma emitters, I-131, Dissolved and entrained gasses, H-3, Gross alpha Sr-89, Sr-90 Fe-55	5.OE-7 (Principle Gamma Emitters) 1.0E-6 (I-131, Fe-55) 1.0E-7 (Gross Alpha) 5.0E-8 (Sr-89, Sr-90) 1.0E-5 (dissolved and entrained noble gases, H-3)	Secondary effluent monitoring
Waste Water Holdup Tanks (WWHT)	Per ODCM	Liquid Grab Sample	1000	Negligible but can vary	Principal gamma emitters, I-131, Dissolved and entrained gasses, H-3, Gross alpha Sr-89, Sr-90 Fe-55	5.OE-7 (Principle Gamma Emitters) 1.0E-6 (I-131, Fe-55) 1.0E-7 (Gross Alpha) 5.0E-8 (Sr-89, Sr-90) 1.0E-5 (dissolved and entrained noble gases, H-3)	Secondary effluent monitoring
Downstream of SG Blowdown Demineralizers	As Required	Liquid Grab Sample	1000	Will vary	Gamma Spec.	N.A.	To evaluate performance of demineralizers

11A COMPLIANCE WITH APPENDIX I

The dose evaluation presented in this Appendix was developed in support of the original license. These release and dose estimates are considered historical and not subject to future updating. This information is retained to avoid loss of original design basis.

The Offsite Dose Calculation Manual (ODCM) provides guidance requirements for system operation, dose calculations, and monitoring requirements to ensure CPNPP compliance with effluent limits. Actual measured concentrations of radioactivity released and real time dilution and dispersion estimates are utilized to verify compliance with effluent limits. Therefore, CPNPP operation within the requirements of the ODCM ensures compliance within effluent limits, rather than operations within the nominal assumptions utilized in the dose evaluation presented in this Chapter. Actual radioactivity release quantities and associated doses to the public can be found in the annual radioactive effluent release reports submitted to the NRC.

11A.1 CRITICAL PATHWAYS

Pathways of human exposure to plant radioactive emissions likely to account for most of the exposure from plant operation are discussed in this section. These discussions are based upon a knowledge of the characteristics of the site environment and on predicted effluent releases presented in Tables 11A-1 and 11A-4.

These analyses show compliance with 10CFR50 Appendix I design objectives with respect to reactor siting and obtaining an operating license for CPNPP Units 1 and 2. The analyses and discussions are considered complete except for potential future updates that may be required to reflect specific source term changes due to plant design modifications or changes in operating practices. Changes in the local site environment will normally be updated per routine annual land use census surveys according to the methodology and annual reporting requirements of the CPNPP Offsite Dose Calculation Manual (ODCM). Since the CPNPP ODCM effectively monitors Luminant Power's continued compliance with 10CFR50, Appendix I radioactive effluent limits, dynamic revision of this FSAR section due to local site environmental changes is not warranted.

A schematic depicting generalized potential pathways of human exposure to radionuclides in plant effluents is presented in Figure 11A-1, which has been taken from Appendix H of Regulatory Guide 4.2[1]. Studies of current and projected land and water use in the area around the Comanche Peak Nuclear Power Plant have revealed that some pathways shown in Figure 11A-1 do not present significant, potential pathways of exposure from plant effluents.

The cultivated land in the area of the site is primarily peanut fields and improved pasture, with some acreage in commercial vegetables such as squash, okra, and some tomatoes and cucumbers. Minor use is made of Lake Granbury waters for the irrigation of croplands [2]. By far the greatest amount of cultivated land is devoted to livestock production, of which beef production is most important. Ingestion of goat milk or crops actually cultivated on irrigated land are not considered significant pathways in evaluating the radiological impact of the plant. As is normally the case with nuclear plants located on or near fresh water bodies, ingestion of invertebrates or aquatic plants harvested from Squaw Creek Reservoir, Lake Granbury, or Lake Whitney does not present a significant exposure pathway to man. Since the water from Squaw Creek Reservoir is not used as a drinking water supply, ingestion of drinking water from the reservoir is not considered to be an exposure pathway.

The relative importance of the remainder of the potential pathways to man has been evaluated by calculating estimated doses from routine operation of the Comanche Peak Nuclear Power Plant for each pathway. The assumptions, methodology and results of the evaluation are presented below along with the summarized dose calculation results. Annual liquid and gaseous releases include consideration of an extended fuel cycle. Expected long term concentrations in Squaw Creek Reservoir and the resultant liquid pathway doses for the total body, liver, and GI-LLI are considered to be primarily affected by the longer fuel cycle and, therefore, have been evaluated numerically.

Examination of Table 11A-2 reveals that, based upon the dose calculation assumptions described above, the liquid pathway of primary importance in individual total-body exposure will probably be ingestion of fish caught in Squaw Creek Reservoir. Exposure from shoreline activity will generally be of less importance, and swimming and boating pathways would contribute only a minor amount to the dose from plant liquid effluents. The crop irrigation pathway is generally minor, except if Squaw Creek Reservoir water actually was to be used. Results of the dose calculations suggest that the relative importance of liquid pathways is about the same for individual organs as for total-body exposure. The relative importance of each liquid pathway is shown in Table 11A-2 for each of the three reservoirs (Squaw Creek Reservoir, Lake Granbury, and Lake Whitney).

A summary of the gaseous pathway dose calculation results which appears in Table 11A-3 indicates that the vegetable pathway produces the greatest total-body doses and the cow milk pathway yields the maximum organ dose to the thyroid gland of an infant. Cloud submersion, air inhalation, and meat consumption produce smaller contributions to the total dose; irradiation from ground-plane disposition of radionuclides in the plume is the least important total-body exposure pathway.

11A.2 DISPERSION OF NORMAL RADIONUCLIDE RELEASES INTO SURFACE WATERS

During normal operation of the plant, some radionuclide elements will be released into the circulated cooling water. The dispersion of these radionuclides in the surface water environment and an estimation of their maximum concentrations are discussed below.

The released radionuclides will enter the Squaw Creek Reservoir and follow the flow pattern shown in Figure 11A-2. Normally, only a small amount (typically 1759 acre-feet/year) of water will be released from Squaw Creek Reservoir into Squaw Creek. For blow-down purposes, water may be pumped from the reservoir to Lake Granbury, which is a lake created by the De Cordova Bend Dam built on the Brazos River. To keep the volume of Squaw Creek Reservoir constant, a typical volume of 27,900 acre-feet of water per year may be pumped from Lake Granbury into the Squaw Creek Reservoir to replace evaporation and blow-down losses. A portion of this water may be diverted into Lower Squaw Creek. The make-up water to Lake Granbury is provided by the Brazos River. Water released from the De Cordova Ben Dam will flow down the Brazos River 80 miles to Whitney Reservoir.

In order to calculate radionuclide concentrations due to normal releases in the surface water bodies a mathematical model was developed based upon mass balances using a modification of models found in NRC Reg. Guide 1.113[3]. The sketch of the model is presented in Figure 11A-3. The flow in Lake Granbury and Whitney Reservoir was assumed to be plug flow

and the flow in Squaw Creek Reservoir was assumed to be completely mixed. All of the reservoir volumes were assumed to be constant.

The terms used in developing the model are defined below:

C_0 = radionuclide concentration in the Brazos River upstream from Lake Granbury

C_1 = radionuclide concentration flowing into Squaw Creek Reservoir

= C_3 (for conservatism)

C_2 = radionuclide concentration in Squaw Creek Reservoir

C_3 = radionuclide concentration in Lake Granbury

C_4 = radionuclide concentration flowing into Lake Granbury

C_5 = radionuclide concentration of Whitney Reservoir

C_6 = inflow radionuclide concentration into Whitney Reservoir

q_b = pumpage rate to Lake Granbury

q_s = flow rate from Squaw Creek Reservoir to Squaw Creek

Q_B = Annual average flow rate in the Brazos River downstream from Lake Granbury

Q_{Bo} = annual average flow rate in the Brazos River downstream from Whitney Reservoir

Q_L = pumpage rate from Lake Granbury

Q_{EG} = evaporation from Lake Granbury

Q_{ES} = evaporation from Squaw Creek Reservoir

Q_{EW} = evaporation from Whitney Reservoir

$t_{1/2}$ = half-life of the radionuclide considered

V_G = volume of Lake Granbury

V_S = volume of Squaw Creek Reservoir

V_W = volume of Whitney Reservoir

W = rate of release of the radionuclide considered

For calculation purposes, the evaporation from Squaw Creek Reservoir is assumed to equal (Q_L-q_b-q_s). Evaporation in Lake Granbury averages 58 inches per year (Climatic Atlas of the United States, U.S. Department of Commerce, June 1969). The constant average volume of Lake Granbury is assumed to be 83,000 acre-feet, which is the average of the top gate capacity of 150,000 acre-feet and crest capacity of 15,000 acre-feet. The corresponding surface area for 83,000 acre-feet is about 5,000 acres. This area yields about 24,000 acre-feet of evaporation loss per year. Evaporation in Whitney Reservoir averages 54 inches and for calculation purposes, is equal to (Q_B-Q_{Bo}+q_s).

Using the complete mixing model, the concentration in Squaw Creek Reservoir can be found as shown below:

$$C_2 = \frac{(C_3 - Q_L) + W}{q_s + q_b + \left(\frac{V_s \cdot 1n2}{t_{1/2}}\right)} \qquad 11A\text{-}1$$

For the case of tritiated water only, an evaporation term for Squaw Creek Reservoir (Q_{ES}) is included:

$$C_2 = \frac{(C_3 - Q_L) + W}{Q_{ES} + q_s + q_b + \left(\frac{V_s - 1n2}{t_{1/2}}\right)} \qquad 11A\text{-}2$$

The concentration balance equation for water entering Lake Granbury is:

$$C_0 (Q_B + Q_L + Q_{Eg}) + C_{2qb} = (Q_B + Q_L + Q_{Eg}) C_4 \qquad 11A\text{-}3$$

$$C_0 = 0 \text{ (fresh makeup water)} \qquad 11A\text{-}3a$$

Using the plug flow formula, the concentrations in Lake Granbury and Whitney Reservoir can be found as shown:

$$C_3 = \exp\left[-\left(\frac{V_G \cdot 1n2}{t_{1/2} \cdot (Q_B + Q_L)}\right)\right] C_4 \qquad 11A\text{-}4$$

$$C_5 = \exp\left[-\left(\frac{V_w \cdot 1n2}{t_{1/2} \cdot Q_{Bo}}\right)\right] C_6 \qquad 11A\text{-}5$$

where $Q_{Bo} = Q_B + q_s$- Evaporation from Whitney Reservoir

Concentration Balance of water into Lake Whitney:

$$C_6(Q_B + q_s) = C_2q_s + C_3Q_B \qquad \text{11A-6}$$

The five unknowns (C_2, C_3, C_4, C_5, C_6) can be solved for from the equations presented above.

In the mathematical model above, the following constants were used:

q_s	=	1,759 ac-ft/yr
q_b	=	24,900 ac-ft/yr (for Lake Granbury and Whitney Reservoir Concentration Calculations)
	=	0 ac-ft/yr (for Squaw Creek Reservoir Concentration Calculation)
V_s	=	151,000 ac-ft
V_G	=	83,000 ac-ft
Q_B	=	1,000,000 ac-ft/yr
V_w	=	627,000 ac-ft
Q_{Bo}	=	994,000 ac-ft/yr

The steady state concentrations for each raionuclide element subject to normal releases were calculated using the mathematical model discussed above. The results for Squaw Creek Reservoir, Lake Granbury, and Whitney Reservoir are presented in Table 11A-1.

11A.3 ESTIMATED DOSES FROM LIQUID EFFLUENTS

Doses to individuals in the environs of the plant were calculated for all potentially significant liquid pathways. Doses to individuals were calculated for drinking water, aquatic food consumption, recreational activity and crop irrigation pathways, as applicable to Squaw Creek Reservoir, Lake Granbury, or Whitney Reservoir. Assumptions, including point of exposure, are described for each pathway in the following paragraphs; the calculated liquid pathway doses are summarized in Table 11A-2. Release values presented in Table 11A-1, which reflect two unit operation, were used. All results were based upon the computational techniques presented in Reg. Guide 1.109[4]. All usage and consumption values, transport times, bioaccumulation factors, dose conversion factors, and other constants utilized were those suggested in Reg. Guide 1.109 and Ref. [9]. The midpoint of plant life is 20 years, in accordance with Ref. [9].

The nearest downsteam municipal use of Brazos River water is at Waco, nearly 140 miles downstream of Lake Granbury. The Brazos River Water Authority has an application for 10,000 acre-feet of water from Lake Granbury for municipal use. At present, however, the city of Marlin (located 170 miles downstream of Lake Granbury) is the nearest municipal water user who has contracted to purchase water from the Brazos River Authority. Nevertheless, the dose

to an individual obtaining his entire annual water requirement from either Whitney Reservoir or Lake Granbury was estimated. It was assumed that 12 hours would elapse from the time of withdrawal to uptake by the individual. The maximum calculated dose to a single organ from this pathway was 7.14×10^{-2} mrem/yr to a child's GI-LLI; the maximum whole-body dose was 6.92×10^{-2} mrem/yr to a child. Drinking water doses at actual points of withdrawal are expected to be lower than those presented due to additional dilution and decay in transit.

Radionuclides released from the plant were assumed to be immediately available for uptake by aquatic food. Radionuclide concentrations in aquatic food were calculated on the assumption that they are in equilibrium with average concentrations in the respective reservoir where they reside after buildup during the assumed plant lifetime of forty years. The maximum predicted dose to a single organ from the fish consumption pathway was due to the fish from Squaw Creek Reservoir, 5.07 mrem/yr to a teen's liver. The maximum whole-body dose was 3.47 mrem/yr to an adult eating fish from Squaw Creek Reservoir.

Exposure to an individual engaging in recreational activity on the shore was evaluated with the 40 year equilibrium concentrations assumed the same at every point about the circumference of the particular reservoir. The doses from shoreline exposure are summarized in Table 11A-2. The maximum predicted total dose to a single organ from recreational activity at Squaw Creek Reservoir was 8.82×10^{-2} mrem/yr to the skin of a teen. The maximum calculated teen total-body dose was 7.52×10^{-2} mrem/yr.

Conservative assumption regarding the contaminated food pathway were made in order to evaluate this pathway in accordance with Appendix A, paragraph 2d of Reg. Guide 1.109. It was assumed that an individual was exposed to vegetation that was irrigated with contaminated water, and exposed to milk and meat from animals which consumed contaminated food and water. The contaminated water was assumed to originate from Lake Granbury or Whitney Reservoir, and the concentration of radionuclides existing in each reservoir was the same concentration used for irrigation and consumption by animals.

An average irrigation rate of 0.1 liter/square-meter/hour was used. The total dose to an individual was based upon ingestion of produce, leafy vegetables, goat-milk, and meat. The largest expected total body dose was 1.91×10^{-1} mrem/yr to a child influenced by water in Whitney Reservoir, and the maximum organ dose was 3.06×10^{-1} mrem/yr to the GI-LLI of an adult based on water from Lake Granbury. Doses for both reservoirs are shown in Table 11A-2. The calculated doses were still quite low considering the conservative assumptions used.

The maximum individuals doses calculated as described above were used to evaluate the status of conformance of predicted liquid effluents from the Comanche Peak Nuclear Power Plant with the requirements of Appendix I of 10 CFR Part 50. The assumptions and results of this evaluation are summarized in Table 11A-5. Note that the calculated doses indicate that liquid effluents from the plant conform to the "as low as reasonably achievable" criteria established in paragraphs IIA, IIB, and IIC of Appendix I [5].

11A.4 ESTIMATED DOSES FROM GASEOUS EFFLUENTS

Potential pathways of exposure of man to radionuclides in gaseous effluents from the CPNPP were identified and discussed previously. Dose to individuals in the environs of the plant were

calculated for all potentially significant pathways. The results of the calculations and the assumptions and methodology are described in the following sections.

All results were obtained based upon the computational techniques presented in Reg. Guide 1.109. Except in the case of an eight-month cow grazing season [7] and use of a 20 year midpoint of plant life [9], all usage and consumption values, transport times, bioaccumulation factors, dose conversion factors, and other constants utilized were those suggested in Reg. Guide 1.109.

Dilution factors for atmospheric pathways were calculated according to methods prescribed in Reg. Guide 1.111[8], as discussed in Chapter 11.

Maximum estimated doses to individuals were calculated for cloud submersion, ground plane contamination, inhalation, and cow's milk, goat's milk, vegetable, and meat ingestion pathways. Assumptions, including point of exposure, are described for each pathway in the following paragraphs; the calculated gaseous pathway doses are summarized in Table 11A-3. All estimates were based upon predicted gaseous releases in Table 11A-4. Each dose was calculated at the location of the highest dose offsite at which the pathway could be assumed to exist.

Exposure to an individual from submersion in a cloud containing radioactive noble gas effluents was evaluated at the exclusion boundary with the largest atmospheric dilution, 1.29 miles in the north-northwest sector, and at two residences, one 1.55 miles west and the other 1.95 miles west-northwest of the plant. The estimated total-body doses for the three locations were 1.52×10^{-1} mrem/yr, 3.59×10^{-2} mrem/yr, and 3.54×10^{-2} mrem/yr, respectively; corresponding skin doses were 3.91×10^{-1} mrem/yr, 9.23×10^{-2} mrem/yr, and 9.11×10^{-2} mrem/yr, respectively.

External irradiation from activity deposited on ground surfaces was also evaluated at the exclusion area boundary point with maximum concentrations as well as at the two above-mentioned most important residences. At the north-northwest boundary, 2.81×10^{-2} mrem/yr to the total-body and 3.29×10^{-2} mrem/yr to the skin can be expected annually. An individual at the residence 1.55 miles west of the plant will experience estimated doses of 3.83×10^{-3} mrem/yr to the total body and 4.49×10^{-3} mrem/yr to the skin, respectively, while at the residence 1.95 miles west-northwest of the plant, 4.11×10^{-3} mrem/yr to the total body and 4.82×10^{-3} to the skin are the estimated levels.

The maximum individual dose from the air inhalation pathway is reported at the same three locations at which plume submersion and ground deposition doses were reported. The maximum organ dose to an individual at these locations is expected to a teen's thyroid gland.

Inhalation of tritium causes about 79 percent of the thyroid dose. The adult thyroid doses are estimated as 4.73×10^{-1} mrem/yr, 1.12×10^{-1} mrem/yr, and 1.10×10^{-1} mrem/yr at the north-northwest site boundary, the residence at 1.55 miles west, and the residence at 1.95 miles west-northwest of the plant, respectively.

The predicted dose to an individual obtaining 76 percent (stipulated in Regulatory Guide 1.109) of his vegetable (including leafy vegetable) consumption from a garden adjacent to the two residences was estimated. Gardens were assumed to be located at 1.55 miles west and

1.89 miles west-northwest and correspond to the two above-mentioned residences. The maximum calculated exposure from this pathway was to a child's bone, 1.41×10^{-1} mrem/yr 1.55 miles west of the plant and 1.48×10^{-1} mrem/yr 1.95 miles west-northwest of the plant. Corresponding total-body doses at these locations were 5.62×10^{-1} and 5.91×10^{-1} mrem/yr. Approximately 99 percent of the bone dose and 50 percent of the total body dose was due to C-14.

For eight months of the year, cows and goats are expected to graze on pasture land exposed to radioparticulates deposited from the CPNPP effluent plume. Since an objective of this study was to determine the maximum point of exposure for each pathway, the meteorological diffusion conditions were used to select the highest dose location. It has been determined that two milk cows are used for home use and feeding calves at 1.55 miles in the west sector. In addition, two Holstein cows are reportedly milked for home use 1.89 miles west- northwest of the plant. The cow-milk pathway was evaluated at both locations since the resultant thyroid doses were deemed to be of the same magnitude. The maximum organ dose from ingestion of milk from a cow grazing eight months of the year at the two locations mentioned above occurred to an infant's thyroid gland and was estimated as 1.51 mrem/yr and 1.70 mrem/yr, respectively. It is postulated that the infant will receive the maximum total-body dose at those locations, 4.44×10^{-1} mrem/yr and 4.67×10^{-1} mrem/yr, respectively. Seventy percent of these thyroid doses are attributed to I-131, as expected, while 62 percent of the total body dose came from C-14.

There are indications that domestic goats may be used locally for milk in the vicinity of the plant. Therefore, consumption of goat's milk is considered an expected pathway to man during the life of the plant, even though there is no indication of the sale of goat's milk commercially. Five goats are milked locally at a site 4.18 miles north of the plant. In addition, about 30 brush goats are located in the same wind sector 3.37 miles from the plant. While it is remote that an individual will catch and milk these brush goats, the possibility cannot be ruled out. Therefore, both the milk goats at 4.18 miles north and the brush goats at 3.37 miles north were evaluated for the goat's milk pathway. It was estimated that the maximum organ dose is again to an infant's thyroid 8.17×10^{-1} mrem/yr from the milk goats at 4.18 miles north and 1.29 mrem/yr from the brush goats. The respective maximum total-body doses were to an infant at 2.14×10^{-1} mrem/yr and 3.10×10^{-1} mrem/yr.

Exposure from consumption of meat was evaluated at a site 2.12 miles east-southeast of the plant where there are yearlings among beef cattle but no milk cows. The maximum organ dose to an individual from ingestion of meat from cattle grazing eight months of the year was to a child's bone at this location, 1.14×10^{-1} mrem/yr. The maximum total body dose from the meat ingestion pathway was estimated as 3.15×10^{-2} mrem/yr.

Maximum individual doses calculated as described above were used to evaluate the status of conformance of predicted gaseous effluents from the CPNPP with the requirements of Appendix I [5]. The assumptions and results of this evaluation are summarized in Table 11A-6. Beta and gamma air doses were calculated in accordance with the methods stated in Reg. Guide 1.109[4]. It will be noted that the calculated doses indicate that the plant design conforms to the "as low as reasonably achievable" criteria established in paragraphs IIA, IIB, and IIC of Appendix I [5].

REFERENCES

1. Regulatory Guide 4.2, Revision 2, Preparation of Environmental Reports for Nuclear Power Stations, July 1976.

2. Texas Water Rights Commission, 1977, Personal Communications.

3. Regulatory Guide 1.113, Revision 1, Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I, April 1977.

4. Regulatory Guide 1.109, Revision 1, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, October 1977.

5. Title 10, Code of Federal Regulations, Part 50, Appendix I.

6. Deleted.

7. Final Environmental Statement, Comanche Peak Steam Electric Station Units 1 and 2, Texas Utilities Generating Company, Docket Nos. 50-445 and 50-446, United States Atomic Energy Commission (June 1974) p. 5-29.

8. Regulatory Guide 1.111, Revision 1, methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, July 1977.

9. NUREG/CR-4013 “LADTAP II - Technical Reference and User Guide,” April 1986.

TABLE 11A-1
EXPECTED CONCENTRATIONS[b] OF RADIOACTIVE MATERIALS IN ENVIRONMENTAL MEDIA FROM LIQUID EFFLUENTS OF THE COMANCHE PEAK NUCLEAR POWER PLANT - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 3)

Isotope	Annual[b] Release (Ci/yr)	Expected Long[c] Term Concentration of Squaw Creek Reservoir uCi/ml	Expected Long[c] Term Concentration of Lake Granbury uCi/ml	Expected Long[c] Term Concentration of Whitney Reservoir uCi/ml
H-3	5.19E+03	9.40E-05	6.39E-07	6.62E-07
CR-51	8.40E-02	4.95E-11	5.34E-13	1.96E-15
MN-54	4.60E-02	2.92E-10	4.90E-12	3.12E-12
FE-55	3.40E-02	6.56E-10	8.69E-12	7.90E-12
FE-59	8.40E-03	8.00E-12	1.10E-13	3.52E-15
CO-58	1.30E-01	1.96E-10	3.07E-12	3.59E-13
CO-60	1.56E-02	5.84E-10	5.83E-12	5.76E-12
ZR-95	1.10E-02	1.51E-11	2.32E-13	2.19E-14
NB-95	8.20E-03	6.08E-12	7.48E-14	8.81E-16
NP-239	3.40E-02	1.69E-12	1.54E-17	[d]
SR-89	4.00E-03	4.40E-12	6.35E-14	3.26E-15
MO-99	1.08E-01	6.39E-12	1.97E-16	[d]
TC-99M	1.02E-01	5.41E-13	[d]	[d]
RU-106	4.86E-01	3.72E-09	3.27E-11	2.28E-11
AG-11OM	3.80E-02	2.02E-10	3.43E-12	1.97E-12
TE-129M	5.20E-03	3.74E-12	4.54E-14	4.68E-16
I -131	1.06E-01	1.81E-11	3.96E-14	1.68E-22
TE-132	3.00E-02	2.07E-12	1.61E-16	[d]
I -132	3.00E-02	6.10E-14	[d]	[d]

TABLE 11A-1

EXPECTED CONCENTRATIONS[b] OF RADIOACTIVE MATERIALS IN ENVIRONMENTAL MEDIA FROM LIQUID EFFLUENTS OF THE COMANCHE PEAK NUCLEAR POWER PLANT - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 3)

Isotope	Annual[b] Release (Ci/yr)	Expected Long[c] Term Concentration of Squaw Creek Reservoir uCi/ml	Expected Long[c] Term Concentration of Lake Granbury uCi/ml	Expected Long[c] Term Concentration of Whitney Reservoir uCi/ml
I- 133	1.06E-01	1.97E-12	3.79E-23	[d]
CS-134	6.46E-03	9.92E-11	4.81E-13	4.18E-13
I -135	3.40E-02	2.01E-13	[d]	[d]
CS-136	2.40E-04	6.61E-14	3.42E-16	2.08E-21
CS-137	5.72E-03	8.86E-10	1.77E-12	1.87E-12
CE-144	1.26E-01	7.51E-10	1.12E-11	6.92E-12
NA- 24	2.40E-01	3.18E-12	1.48E-26	[d]
NI- 63	4.00E-05	1.12E-11	2.57E-14	2.75E-14
ZN- 65	1.46E-02	7.50E-11	1.28E-12	7.21E-13
W -187	2.20E-02	4.65E-13	1.12E-22	[d]
SR- 90	5.66E-04	8.39E-11	1.99E-13	2.10E-13
Y - 90	1.60E-04	9.06E-15	2.08E-19	[d]
SR- 91	2.40E-03	2.05E-14	[d]	[d]
Y - 91M	1.62E-03	1.19E-15	[d]	[d]
Y - 91	3.20E-04	3.98E-13	5.96E-15	4.35E-16
Y - 93	1.20E-02	1.08E-13	[d]	[d]
NB-95M	8.00E-05	6.36E-15	1.04E-18	[d]
RU-103	2.18E-01	1.83E-10	2.19E-12	4.34E-14
RH-103M	2.00E-01	1.68E-13	[d]	[d]

TABLE 11A-1

EXPECTED CONCENTRATIONS[b] OF RADIOACTIVE MATERIALS IN ENVIRONMENTAL MEDIA FROM LIQUID EFFLUENTS OF THE COMANCHE PEAK NUCLEAR POWER PLANT - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 3)

Isotope	Annual[b] Release (Ci/yr)	Expected Long[c] Term Concentration of Squaw Creek Reservoir uCi/ml	Expected Long[c] Term Concentration of Lake Granbury uCi/ml	Expected Long[c] Term Concentration of Whitney Reservoir uCi/ml
TE-129	3.40E-03	3.45E-15	[d]	[d]
TE-131M	1.54E-02	4.08E-13	4.04E-21	[d]
TE-131	2.80E-03	1.03E-15	[d]	[d]
I -134	6.00E-05	4.67E-17	[d]	[d]
BA-137M	2.60E-03	9.76E-17	[d]	[d]
BA-140	3.20E-01	8.68E-11	4.39E-13	2.22E-18
LA-140	5.00E-01	1.78E-11	7.36E-18	[d]
CE-141	4.00E-03	2.75E-12	3.24-14	2.66E-16
CE-143	3.20E-02	9.37E-13	3.67E-20	[d]
PR-143	4.40E-03	1.28E-12	7.09E-15	7.95E-20
PR-144	1.12E-01	2.85E-14	[d]	[d]

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

b) per two reactor site based on extended fuel cycle

c) 40 year maximum equilibrium concentration (Squaw Creek Reservoir concentration based on extended fuel cycle)

d) negligible concentration after 40 years

TABLE 11A-2
SUMMARY OF CALCULATED LIQUID PATHWAY DOSES[a] - ORIGINAL LICENSING BASIS[b]

(Sheet 1 of 3)

Pathway	Location	Age Group	Organ Receiving Maximum Dose: Organ	Organ Receiving Maximum Dose: Dose (mrem/yr)	Total-Body Dose (mrem/yr)
Drinking Water	Lake Granbury	Adult	GI-LLI	5.51E-02	4.94E-02
		Teen	GI-LLI	3.90E-02	3.48E-02
		Child	GI-LLI	7.05E-02	6.68E-02
		Infant	GI-LLI	6.78E-02	6.54E-02
Drinking Water	Whitney Reservoir	Adult	GI-LLI	5.48E-02	5.12E-02
		Teen	GI-LLI	3.88E-02	3.61E-02
		Child	GI-LLI	7. 14E-02	6.92E-02
		Infant	GI-LLI	6.91E-02	6.77E-02
Fish Ingestion	Squaw Creek Reservoir	Adult	Liver	4.93E+00	3.47E+00
		Teen	Liver	5.07E+00	2.02E+00
		Child	Liver	4.53E+00	9.00E-01
Aquatic Food Ingestion	Lake Granbury	Adult	GI-LLI	5.63E-02	1.36E-02
		Teen	GI-LLI	4.02E-02	9.43E-03
		Child	Liver	2.24E-02	6.48E-03
Aquatic Food Ingestion	Whitney Reservoir	Adult	GI-LLI	3.63E-02	1.26E-02
		Teen	GI-LLI	2.60E-02	8.34E-03
		Child	Liver	1.90E-02	5.30E-03

TABLE 11A-2
SUMMARY OF CALCULATED LIQUID PATHWAY DOSES[a] - ORIGINAL LICENSING BASIS[b]

(Sheet 2 of 3)

Pathway	Location	Age Group	Organ Receiving Maximum Dose: Organ	Organ Receiving Maximum Dose: Dose (mrem/yr)	Total-Body Dose (mrem/yr)
Shoreline Activity	Squaw Creek Reservoir	Adult	Skin	1.58E-02	1.34E-02
		Teen	Skin	8.82E-02	7.52E-02
		Child	Skin	1.84E-02	1.57E-02
Shoreline Activity	Lake Granbury	Adult	Skin	1.09E-04	9.27E-05
		Teen	Skin	6.09E-04	5.17E-04
		Child	Skin	1.27E-04	1.09E-04
Shoreline Activity	Whitney Reservoir	Adult	Skin	1.01E-04	8.59E-05
		Teen	Skin	5.64E-04	4.79E-04
		Child	Skin	1.17E-04	1.00E-04
Crop Irrigation	Lake Granbury	Adult	Liver	3.06E-01	1.02E-01
		Teen	Liver	2.61E-01	1.20E-01
		Child	Liver	2.05E-01	1.85E-01
		Infant	Liver	1.67E-01	1.56E-01
Crop Irrigation	Whitney Reservoir	Adult	GI-LLI	2.45E-01	1.05E-01
		Teen	GI-LLI	2.20E-01	1.24E-01
		Child	GI-LLI	2.51E-01	1.91E-01
		Infant	Liver	1.73E-01	1.61E-01

TABLE 11A-2
SUMMARY OF CALCULATED LIQUID PATHWAY DOSES[a] - ORIGINAL LICENSING BASIS[b]

(Sheet 3 of 3)

Pathway	Location	Age Group	Organ Receiving Maximum Dose: Organ	Organ Receiving Maximum Dose: Dose (mrem/yr)	Total-Body Dose (mrem/yr)
Milk-cow/Meat	Squaw Creek Reservoir	Adult	GI-LLI	3.98E+00	2.52E+00
		Teen	GI-LLI	3.71E+00	2.80E+00
		Child	GI-LLI	4.82E+00	4.28E+00
		Infant	Liver	5.88E+00	5.76E+00

a) Pathways from Squaw Creek Reservoir to total body, liver, and GI-LLI based on extended fuel cycle.

b) Historical, not subject to future updating. Has been retained to preserve original design basis.

TABLE 11A-3
SUMMARY OF CALCULATED GASEOUS PATHWAY DOSES - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 2)

Pathway	Location	Age Group	Organ Receiving Maximum Dose: Organ	Organ Receiving Maximum Dose: Dose (mrem/yr)	Total-Body Dose (mrem/yr)
Cloud Submersion	Site Boundary (1.29 miles NNW)	All	Skin	3.91E-01	1.52E-01
Ground Plane	Site Boundary (1.29 miles NNW)	All	Skin	3.29E-02	2.81E-02
Contamination Air Inhalation	Site Boundary (1.29 miles NNW)	Adult	Thyroid	4.50E-01	3.76E-01
		Teen	Thyroid	4.73E-01	3.81E-01
		Child	Thyroid	4.47E-01	3.41E-01
		Infant	Thyroid	2.95E-01	1.98E-01
Vegetable Ingestion	Garden (1.89 Miles WNW)	Adult	Thyroid	4.37E-01	2.43E-01
		Teen	Bone	6.14E-01	3.14E-01
		Child	Bone	1.48E+00	5.91E-01
Cows' Milk Ingestion	1.89 Miles WNW	Adult	Thyroid	2.49E-01	8.79E-02
		Teen	Thyroid	3.87E-01	1.31E-01
		Child	Thyroid	7.63E-01	2.55E-01
		Infant	Thyroid	1.70E-00	4.67E-01

TABLE 11A-3
SUMMARY OF CALCULATED GASEOUS PATHWAY DOSES - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 2)

Pathway	Location	Age Group	Organ Receiving Maximum Dose: Organ	Organ Receiving Maximum Dose: Dose (mrem/yr)	Total-Body Dose (mrem/yr)
Goats' Milk Ingestion	Brush Goats 3.37 Miles N	Adult	Thyroid	1.98E-01	7.13E-02
		Teen	Thyroid	3.01E-01	1.00E-01
		Child	Thyroid	5.81E-01	1.80E-01
		Infant	Thyroid	1.29E-00	3.10E-01
Meat Ingestion	2.12 Miles ESE	Adult	Bone	7.16E-02	2.66E-02
		Teen	Bone	6.05E-02	1.94E-02
		Child	Bone	1.14E-01	3.15E-02

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

TABLE 11A-4
EXPECTED MAXIMUM OFF-SITE CONCENTRATIONS[a] OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - ORIGINAL LICENSING BASIS[b]

Isotope	Annual [c] Release (Ci/Yr)	(Expected) [d] Exclusion Area Boundary Concentration (Ci/ml)
H^3	2.80E+03	2.93E-10
Kr^{83m}	2.00E+00	2.09E-13
Kr^{85m}	3.00E+01	3.14E-12
Kr^{85}	5.00E+02	5.23E-11
Kr^{87}	8.00E+00	8.37E-13
Kr^{88}	4.60E+01	4.81E-12
Xe^{131m}	1.40E+01	1.46E-12
Xe^{133m}	4.20E+01	4.39E-12
Xe^{133}	2.20E+03	2.30E-10
Xe^{135}	1.14E+02	1.19E-11
I^{131}	5.00E-02	5.23E-15
I^{133}	5.20E-02	5.44E-15
Mn^{54}	8.80E-04	9.21E-17
Fe^{59}	3.00E-04	3.14E-17
Co^{58}	3.00E-03	3.14E-16
Co^{60}	1.34E-03	1.40E-16
Sr^{89}	6.60E-05	6.91E-18
Sr^{90}	1.18E-05	1.23E-18
Cs^{134}	8.80E-04	9.21E-17
Cs^{137}	1.48E-03	1.55E-16
C^{14}	1.60E+01	1.67E-12
Ar^{41}	5.00E+01	5.23E-12

a) Per two reactor site

b) Historical, not subject to future updating. Has been retained to preserve original design basis.

c) No radioactive decay assumed.

d) Expected concentration in worst sector averaged over a one-year period.

TABLE 11A-5

APPENDIX I CONFORMANCE SUMMARY TABLE FOR LIQUID EFFLUENTS - ORIGINAL LICENSING BASIS[a]

	Appendix I Criteria		Comanche Peak Nuclear Power Plant[b]	
Type of Dose	Design[c] Objective	Point of Dose Evaluation	Calculated Dose	Point of Dose Evaluation
Liquid Effluents				
Dose to total body from all pathways	6 mrem/yr per site	Location of the highest dose offsite	5.99 mrem/yr[d]	Squaw Creek Reservoir[e]
Dose to any organ from all pathways	20 mrem/yr per site	Same as above	8.87 mrem/yr[f]	Squaw Creek Reservoir[e]

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

b) Points given correspond to points of dose evaluation under Appendix I heading.

c) Design objectives as specified in 10CFR50, Appendix I.

d) Dose to adult from fish ingestion, recreational exposure, and milk-cow and meat from animals drinking water.

e) 40 year equilibrium concentration maximum in Squaw Creek Reservior.

f) Dose to child liver from fish ingestion, recreational exposure, and milk-cow and meat from animals drinking water.

TABLE 11A-6

APPENDIX I CONFORMANCE SUMMARY TABLE FOR GASEOUS EFFLUENTS - ORIGINAL LICENSING BASIS[a]

Appendix I Criteria		Comanche Peak Nuclear Power Plant[b]		
Type of Dose	Design[c] Objective	Point of Dose Evaluation	Calculated Dose	Point of Dose Evaluation
Gaseous Effluents[d]				
Gamma dose in air	20 mrad/yr per site	Location of the highest dose offsite [e]	0.238 mrad/yr	Location of highest annual average concentration at the site boundary (NNW at 1.29 mile)
Beta dose in air	40 mrad/yr per site	Same as above	0.427 mrad/yr	Same as above
Dose to total body	5 mrem/yr per site	Location of the highest dose offsite [f]	0.152 mrem/yr	Same as above
Dose to skin of an individual	15 mrem/yr per site	Same as above	0.391 mrem/yr	Same as above
Dose to any organ from all pathways	30 mrem/yr per site	Location of the [g] highest dose offsite	2.20 mrem/yr[h]	Worst cow and garden[i] (WNW at 1.89 miles)

a) Historical, not subject to future updating. Has been retained to preserve original design basis.

b) Doses due to tritium and C-14 intake from terrestrial food chains are included in this category.

c) Design objectives as specified in 10CFR50, Appendix I

d) Calculated only for noble gases.

e) Evaluated at a location that could be occupied during the term of plant operation.

f) Evaluated at a location that is anticipated to be occupied during plant lifetime or evaluated with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation.

g) Evaluated at a location where an exposure pathway actually exists at time of licensing. However, if the applicant determines design objectives with respect to radioactive iodine on the basis of existing conditions and if potential changes in land and water usage and food pathways could result in exposures in excess of the guideline values given above, the applicant should provide reasonable assurance that a monitoring and surveillance program will be performed to determine: 1) the quantities of radioactive iodine actually released to the atmosphere and deposited relative to those estimated in the determination of design objectives; 2) whether changes in land and water usage and food pathways which would result in individual exposures greater than originally estimated have occurred; and 3) the content of radioactive iodine and foods involved in the changes if and when they occur.

h) Dose to a child's bone primarily from air inhalation, vegetation and cow milk ingestion.

i) Two Holstein cows currently milked at this location for home use.

12.0 RADIATION PROTECTION

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12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

12.1.1 POLICY CONSIDERATIONS

The Luminant Power policy, through guidance from Regulatory Guides 1.8, 8.8, and 8.10 and 10 CFR Part 20, includes a strong commitment to ensure that occupational radiation exposures (ORE) are As Low As Reasonably Achievable (ALARA). Luminant Power will require acceptable radiation protection practices for all applicable station activities, and will provide a qualified and vigilant health physics organization and atmosphere to accomplish this goal. Management also recognizes and will emphasize the importance of each individual's responsibilities to maintain ORE-ALARA.

The Plant Manager (See Section 13.1.1.2.1) has the ultimate responsibility for establishing an ALARA philosophy for all applicable aspects of operating, maintaining, refueling and testing of the station. The operating organizational structure and responsibilities are discussed in Section 13.1.2. The Plant Manager informs station personnel of management's commitment to the ORE-ALARA philosophy and authorizes implementation of ALARA policies by issuing appropriate special orders, policy letters or administrative orders. The Manager, Nuclear Radiation Protection is responsible for implementing the ALARA policy.

The Manager, Nuclear Radiation Protection and his staff prepare or review applicable written procedures, develop dose-saving techniques and methods, provide health physics training and personnel guidance, and vigilantly observe for possible deviations from acceptable radiation protection practices.

12.1.2 DESIGN CONSIDERATIONS

Design considerations to ensure that occupational radiation exposures are as low as reasonably achievable, consistent with benefits received and costs incurred, are the following:

1. Applying experience from past and present engineering designs; cognizant engineers and designers have access to project files of completed projects and can contact personnel assigned to other current projects.

2. Applying experience from operating plants; applicable sources for operating plant experience are the following:

   a. NRC Current Events Reports

   b. NRC Operating Experience Bulletin Information Reports

   An example of "applying experience from operating plants" is the use of a concrete wall with a thickness of 2 ft 6 in. instead of a thin steel shell wall for the Refueling Water Storage Tank (RWST) because it has been found that for an operating plant the surface dose on the steel tank exceeds the upper limit of the radiation zone category for the unrestricted area.

3. Following facility and equipment design practices to reduce radiation level and maintenance; these practices are governed by NRC Regulatory Guide 8.8, Revision 2,

Section C.2, which requires that specific features be incorporated in the engineering design to limit ORE to ALARA. Details are presented in Section 12.3.1.

4. Introducing mechanisms that provide for design review by competent professionals in radiation protection and establishing procedures for continuing facility design review.

5. Providing general and specific guidance to individual designers for implementation of design practices which will reduce radiation levels and time spent for maintenance; this general guidance is provided by the comments generated during the review of design drawings.

6. When the operating experience of the Nuclear Steam Supply System (NSSS) supplier was transmitted to both the utility and the Architect/Engineer, design changes were made that could reduce ORE at a reasonable cost.

The primary method of minimizing and controlling the buildup, transport and deposition of activated corrosion products in the reactor coolant system of the Comanche Peak units is by controlling the materials in contact with the primary coolant system. The cobalt isotopes 58 and 60 are the major contributors to occupational radiation exposure during operation, inspection and maintenance of the plant. These isotopes are produced by activation of nickel-58 and cobalt 59, respectively, deposited on the core during power operation. These materials are deposited on the reactor core surfaces after erosion and/or corrosion of the material from piping, valves, etc. in contact with reactor coolant.

The NSSS materials in contact with the reactor coolant are restricted in the quantity of the trace impurity cobalt-59 in the stainless steel and Inconel. The restrictions are shown below:

Component	Material	Maximum Cobalt Content (Wgt %)
Reactor Internals (non-active region)	SS	0.20
Reactor Internals (active region)	SS	0.12
Reactor Vessel Cladding	SS	0.20
Reactor Coolant Piping	SS	0.20
Reactor Internals Bolting MH	SS	0.25
Reactor Coolant Pumps	SS	0.20
Steam Generator Tubes	Inconel	0.016 (Unit 1) 0.10 (Unit 2)
Fuel (non-active region)	SS	0.12
Fuel (active region)	SS	0.08
Fuel	Inconel	0.10

Fuel	Zircaloy	0.002

The other source of cobalt-59 which can be a major contributor to the total cobalt-59 deposited on the reactor core is Stellite hard surfacing materials. The use of Stellite in the NSSS is limited to the minimum application feasible. It is only used in those cases where other hard-surfacing materials are not determined to be suitable from a reliability standpoint. The Stellite application in the NSSS is given below:

Component	Approximate Surface Area
Reactor Internals	3.23 Ft^2
Reactor Coolant Pump Journals	4.30 Ft^2 per pump
Control Rod Drive Mechanisms	10.76 Ft^2
Reactor Coolant System Valves	2.6 Ft^2

The production of cobalt-58 is limited by minimizing the amount of nickel based alloys in contact with the reactor coolant. The reactor coolant system materials in contact with reactor coolant are stainless steel with the exception of steam generator tubing which is the nickel based alloy Inconel. Inconel is used in this application to increase component reliability thereby reducing inspection and maintenance requirements. This in turn leads to a reduction in total occupational radiation exposures. It is expected that the Model D series steam generators at Comanche Peak will exhibit the expected reliability which will minimize inspection and maintenance requirements.

Other methods of minimizing and controlling the buildup, transport and deposition of actuated corrosion products are under investigation by Westinghouse and others, principally under a significant program in this area funded by the Electric Power Research Institute (EPRI) of Palo Alto, California. One of the promising programs being carried out by Westinghouse under the EPRI program is the control of the deposition and transport by stricter controls on reactor coolant chemistry. This and other areas of Westinghouse investigation was presented to the ACRS Environmental Subcommittee on January 25, 1978. Many of these operational techniques, when proven, can be applied to the Comanche Peak units.

The CPNPP design of the auxiliary systems (BOP systems) incorporates recommendations of WCAP-8872 Section 8.0 with the purpose of maintaining occupational radiation exposures as low as reasonably achievable (ALARA).

12.1.3 OPERATIONAL CONSIDERATIONS

Operational considerations at CPNPP that promote the ALARA philosophy are the determination of the origins of radiation exposures, the proper training of personnel, the preparation of radiation protection procedures, the development of conditions for implementing these procedures, and the formation of a review system to assess the effectiveness of the ALARA philosophy.

The Manager, Nuclear Radiation Protection and his staff working closely with other departments review and study station systems such as the NSSS, the radioactive waste systems, the

Residual Heat Removal System, the Spent Fuel Pool Cooling and Cleanup System, and other systems that collect, store, contain, or transport liquid, gaseous, or solid radioactive material. Objectives are to understand the functional aspects of each system, to identity the origins of radiation exposures in the station, and to know these exposure origins by location, operation, and job category.

The radiation protection procedures, prepared and written by the Manager, Nuclear Radiation Protection and his staff, will emphasize acceptable health physics techniques and methods, and when practical will incorporate guidance from Regulatory Guides 8.8 and 8.10. The information received from other operating nuclear power plants, relating their approach and methods used in solving occupational exposure problems are carefully studied and provide significant assistance in the development of the CPNPP procedures. A description of the radiation protection procedures for CPNPP is located in Section 12.5.3.

The implementation of the ALARA philosophy is directed by the Manager, Nuclear Radiation Protection and is accomplished by two modes. In the review and approval of those plant procedures that include the potential for occupational exposure, appropriate radiation protection procedure may be referenced in the plant procedures to ensure that any occupational exposure is maintained as low as reasonably achievable. Entrance to the restricted areas at CPNPP is controlled by the radiation protection section and requires the issuance of a permit. The description of the permit system is discussed in Section 12.5.3. When the permit is initiated, the work assignment and applicable procedures are listed. As determined on a case-by-case basis, additional radiation protection procedures can be implemented at this time if needed.

The Manager, Nuclear Radiation Protection is responsible for the review of exposure records, investigating not only the individual exposures, but the exposures as classified by job description and job location. Information obtained from this review will be compared with exposure results from past experience and with data obtained from average exposure results from other plants to assess the effectiveness of the ALARA effort at CPNPP.

The Nuclear Oversight organization reviews the radiation protection program for compliance and effectiveness and provides a valuable assistance by this critique in assuring that ORE is ALARA.

The station management will periodically review the exposure records and discuss them with the Manager, Nuclear Radiation Protection This review will seek to identify excessive exposure areas, excessive exposures by job categories, and other exposure trends. They will determine if improvements are needed in plant procedures, radiation protection procedures, or plant equipment; and they will initiate these improvements if they will substantially reduce exposures at a reasonable cost.

In the radiation protection program, Section 12.5.3, there is provided a section for radiation protection training at CPNPP. All occupational workers will be instructed as to the philosophy, why Luminant Power has this philosophy, and how the ORE-ALARA philosophy is implemented at CPNPP.

12.2 RADIATION SOURCES

The radiation source term data provided in this section are the design basis values used for the design of plant shielding. As such they are considered historical and not subject to future updating. This information is retained to avoid loss of original licensing basis. As discussed in Chapter 12.0 compliance with occupational exposure limits and controls is ensured and controlled by compliance with 10CFR20 which is implemented at CPNPP via the Radiation Protection Program and the ALARA Policy.

The parameters used in calculating the reactor coolant inventory at the SPU power level is presented in [New] Table 11.1-1A and the concentrations presented in [New] Table 12.2-3A. (Comment: These tables should be obtained from Westinghouse.)

12.2.1 CONTAINED SOURCES

The radiation source terms used as the basis for shield design calculations and dose evaluations are presented for three general conditions of the reactor plant: full-power operation, shutdown, and postulated design basis accident (DBA).

12.2.1.1 Full-Power Operation Design Sources

12.2.1.1.1 Sources of Radiation Inside the Reactor Containment

The primary sources of contained radiation inside the Reactor Containment Building are the core and the Reactor Coolant System (RCS) equipment and pipes, which contain Nitrogen-16 (N-16), fission products from fuel clad defects (reactor coolant activities are calculated based upon fission products from rods generating one percent of the core thermal power), and activated nuclei and corrosion products. Argon-41 (Ar-41) production in the primary coolant is virtually nonexistent.

Tables 12.2-1 and 12.2-2 list the design basis N-16 concentrations around the reactor coolant loop (RCL) and the pressurizer source activities, respectively. Table 12.2-3 details reactor coolant fission and corrosion product activities. These source data represent equilibrium isotropic concentrations based on a 40-year plant life. Table 12.2-4 presents a four-group summary of the gamma and neutron fluxes at the surface of the primary shield concrete.

12.2.1.1.2 Sources of Radiation in Auxiliary Systems

Shielding thicknesses external to the Containment in the Auxiliary Building are determined by the amounts of radiation sources in the reactor plant systems and the adjacent radiation access requirements. The systems which contain radioactivity during operation and are considered in the shield design are the Chemical and Volume Control System (CVCS), Boron Recycle System (BRS), Waste Processing System (WPS), Steam Generator Blowdown Processing System (SGBPS), and the Condensate Polishing System (CPS).

1. Chemical and Volume Control System

   The CVCS functions primarily to provide continuous purification of reactor coolant water. In addition, two subsystems, the boron thermal regeneration system and the seal water

system, provide control of boron concentration in the reactor coolant and supply gland seal water to reactor coolant pumps, respectively.

The following major equipment of the CVCS contain sources of radiation:

a. Heat Exchangers

   1. Regenerative

   2. Excess letdown

   3. Letdown

   4. Boron thermal regeneration

   5. Seal water

b. Filters

   1. Reactor coolant

   2. Seal water injection

   3. Seal water return

c. Demineralizers

   1. Mixed bed (30 ft^3 of resin)

   2. Cation bed (20 ft^3 of resin)

   3. Boron thermal regeneration (70 ft^3 of resin)

d. Other Tanks and Equipment

   1. Volume control tank

   2. Charging pumps

The sources in the volume control tank are considered to correspond to normal operating levels in the tank and are specified as 160 ft^3 of liquid and 240 ft^3 of vapor. The filters are considered homogeneous cylindrical sources with the following dimensions and compositions:

Filter	Source Compositions (in.)	Source Dimensions (Volume %)

Reactor coolant	radius 3.375, length 19.0	62% air, 38% water
Seal water return	radius 3.375, length 19.0	62% air, 38% water
Seal water injection	radius 1.3, length 20.0	90% air, 10% stainless steel

Table 12.2-5 lists the design volumetric or specific source strengths as a function of gamma energies for the equipment in the CVCS.

Additional heat exchanger data are presented in Table 12.2-6.

2. Boron Recycle System

The BRS reclaims boric acid and primary coolant by demineralization, gas stripping, and evaporation. The following equipment in this system contain sources of radiation during normal operation:

a. Demineralizers

1. Recycle evaporator feed (30 ft^3 of resin)

2. Recycle evaporator condensate (20 ft^3 of resin)

b. Filters

1. Recycle evaporator feed

2. Recycle evaporator condensate

3. Recycle evaporator concentrates

c. Evaporator

1. Recycle evaporator

2. Vent condenser

d. Other Tanks and Equipment

1. Recycle holdup tank

2. Recycle evaporator feed pump

The sources in the recycle holdup tank are based on a vapor volume of 500 ft^3 in a holdup tank for a two-unit system. The filters are considered homogeneous cylindrical sources with the following dimensions and compositions:

Filter	Source Compositions (in.)	Source Dimensions (Volume %)
Evaporator feed	radius 3.375, length 19.0	62% air, 38% water
Evaporator condensate and evaporator concentrates	radius 1.25, length 19.0	30% air, 70% water

The vent condenser portion of the recycle evaporator is considered a homogeneous cylindrical source with a diameter of 8 in., a length of 20 in., and the following composition in volume percents: 30 percent stainless steel, 22 percent water, and 48 percent vapor. The evaporator section is considered as a cylinder with a diameter of 3.5 ft and a length of 9.9 feet. The composition in volume percents is 10 percent stainless steel, 77 percent water, and 13 percent air.

Table 12.2-7 lists the design volumetric sources or specific activities as a function of gamma energy for the equipment in the BRS.

3. Waste Processing System

The WPS is designed to store and process various radioactive byproduct materials of the plant. The WPS cleans and recycles the reactor coolant water, collects floor drains, collects radioactive gases, and processes all radioactive waste materials for safe release or transport.

Design basis source strengths at selected gamma energies for the equipment in the WPS are presented in Section 11.1.

4. Steam Generator Blowdown Processing System

The SGBPS provides filtration and demineralization of steam generator secondary coolant to remove radioactivity which can leak from the primary into the secondary coolant.

The filters are considered as homogeneous cylindrical sources 6.625 in. in diameter and 27.5 in. in length, weighing 25 lb (wet). Each demineralizer has 80 ft^3 of resin and the spent resin storage tank has a volume of 3700 gallons.

Table 12.2-8 presents the volumetric source strengths as a function of gamma energies in the filters, demineralizers and spent resin storage tank in the SGBPS.

12.2.1.2 Shutdown Design Sources

During normal shutdown conditions, the following are the primary sources of radiation:

1. The Residual Heat Removal (RHR) system

2. The Spent Fuel Pool Cooling and Purification (SFPCP) System

3. The reactor core fission decay product and activation sources

4. Corrosion product activity in major components

12.2.1.2.1 Residual Heat Removal Sources

The RHR System is placed into operation within four hours after shutdown to reduce the reactor coolant temperature to approximately 140°F. During this time, the RHR pump and heat exchanger are radioactive. Table 12.2-9 lists the maximum design specific activity in the RHR loop for this mode of operation. RHR heat exchanger dimension data are presented in Table 12.2-6.

12.2.1.2.2 Spent Fuel Pool Cooling and Purification Sources

The spent fuel pool is designed to act as a temporary repository for spent reactor fuel. The SFPCP system maintains proper water temperature and water conditions in this pool during and after refueling.

There are two spent fuel pool demineralizers supplied by Hungerford & Terry. Each spent fuel pool demineralizer has 50 ft^3 of resin. Resin used in these demineralizers are as per manufacturer/Westinghouse recommendation. The filter is considered a homogeneous cylindrical source, 10.25 in. in diameter and 20.5 in. in length, weighing 25 lb (wet). Table 12.2-10 lists the volumetric source strengths for the SFPCP filters and demineralizers.

Weekly samples are pulled to determine spent fuel pool demineralizer decontamination factors, and a surveillance will be performed to monitor differential pressures of the demineralizer and filters. Results of the decontamination factor determination which indicates an inability to control ionic contaminants and differential pressure readings which approach the manufacturer's design recommendations will indicate the need for resin and filter replacement. See Section 9.1.3.4 for a brief description of the chemical analyses and operational surveillance performed when the spent fuel pool is in use.

12.2.1.2.3 Core Shutdown Fission Decay and Activation Sources

The core gamma shutdown sources presented in Table 12.2-11 are based upon an average power assembly with an irradiation time of 10^8 sec, or 3.1 years. These source strengths are tabulated per unit volume of homogenized core material as a function of time after shutdown. The homogenization of the core material is in accordance with the volume fractions presented in Table 12.2-12.

The irradiated control rod sources are used in establishing shielding requirements during refueling operations and during shipping of irradiated rods. The neutron absorber material used in the control rods is silver-indium-cadmium (Ag-In-Cd). The source strengths associated with the control rods are presented in Table 12.2-13 as a function of various times after shutdown. The values are per cm of the height of a single rod for an irradiation period of 100,000 hours.

The in-core detector drive wire sources are used in establishing shielding requirements for the wires when the detectors are not in the core and during shipment when the detectors have failed. Table 12.2-14 lists the detector wire sources per cm of length of wire; it is assumed that the detector has been lodged in the core for one year.

12.2.1.2.4 Corrosion Product Activity in Major Components

Deposited crud activity in major components is a significant source of radiation to personnel in the immediate vicinity of these components. Table 12.2-2 presents deposited crud concentrations in the pressurizer.

The activities on the primary side surfaces of the steam generator are used to determine access limitations in and around the steam generators at plant shutdown. Nominal values of deposited activity are listed in Table 12.2-15 for several operating times.

12.2.1.3 Design Basis Accident Sources

The fission product sources considered released to the Containment following an equivalent 100 percent core meltdown are based on the following assumptions stated in TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites [1]:

1. 100 percent of the noble gases

2. 50 percent of the halogens

3. 1 percent of the remaining fission product inventory

The sources from which the postaccident radiation levels outside the Containment are calculated appear in Table 12.2-16.

The loss of coolant accident (LOCA) is the design basis accident (DBA) used to develop the radiation source terms for equipment qualification and vital area dose evaluation. The post-accident radiation sources of importance are listed in Table 12.2-16, along with the fractions of core activity inventory mixed in the respective fluids. The appropriate source concentrations presented below represent those developed to support the original license. The impact of the Stretch Power Uprate on post-accident radiation environments was developed using source term scaling techniques using the core inventory utilized to support the original license and that developed for the Stretch Power Uprate and the fractional mix presented in Table 12.2-16.

12.2.1.3.1 Containment Atmosphere

The airborne activity within the containment building is the source of radiation to areas outside the containment, in the form of direct radiation through the walls, streaming through penetrations,

and skyshine. The containment atmosphere source terms at various times after a LOCA are given in Table 12.2-17 for Unit 1 and in Table 12.2-17A for Unit 2.

12.2.1.3.2 Containment Atmosphere in the Post-Accident Sampling System

The post accident sampling system, which is located outside the containment building, is not used post accident.

12.2.1.3.3 Primary Coolant

The pressurized LOCA is a postulated LOCA event in which the primary coolant system is conservatively assumed not to depressurize. The systems which may include pressurized primary coolant after a pressurized LOCA include the reactor coolant system and the residual heat removal system. The corresponding primary coolant source terms at various times after a LOCA are provided in Table 12.2-19 for Unit 1 and in Table 12.2-19C for Unit 2.

12.2.1.3.4 Containment Sump Water

The systems outside the containment building which may contain sump water after a depressurized LOCA include the residual heat removal system, the containment spray system, and portions of the safety injection and chemical and volume control systems. The sump water source terms at various times after a LOCA are given in Table 12.2- 19A for Unit 1 and in Table 12.2-19D for Unit 2.

12.2.1.3.5 Post-Accident Sampling System

The post-accident sampling system is not used at CPNPP. Post-accident samples can be obtained using the Process Sampling System (PSS) which may contain either containment atmosphere concentrations described in Section 12.2.1.3.2, containment sump water described in Section 12.2.1.3.4, or primary coolant described in Section 12.2.1.3.3 (with appropriate adjustment for fluid density). The source terms for the latter case at various times after a LOCA are presented in Table 12.2-19B for Unit 1 and in Table 12.2-19E for Unit 2.

12.2.1.3.6 Gap Activity Release

The gap activity is the fraction of core activity that diffuses to the gap between the fuel and cladding. For design basis accidents, the noble gas and iodine inventory in the fuel gas is assumed to be 10 percent of the core noble gas and iodine inventory, except for Kr-85 which is assumed to be 30 percent. The design basis gap activities are presented in Table 15.6-8.

12.2.1.4 Byproduct, Source and Special Nuclear Materials

Information on the projected types, quantities, forms and uses of byproduct, source and special nuclear material as defined in 10 CFR Part 30, 10 CFR Part 40 and 10 CFR Part 70 is provided in Table 12.2-20.

These radioactive materials are handled in accordance with the controls described in Section 12.5.3.7, “Radioactive Materials Control”. Special attention is given to the 5000 Ci Cs-137 source used for calibration of portable radiation detection instrumentation, including high range instrumentation available for performing surveys during emergency conditions. This

source is located in the Radiation Protection Calibration Facility and is housed in a Calibration Well Assembly. The calibration facility, well assembly, shield design, physical barriers and administrative and procedural controls associated with this source are discussed in Section 12.5.2.1.

12.2.1.5 Assumptions, Calculation Methods and Design Basis for Contained Radiation Sources

Radiation shielding is designed to provide radiation protection for plant personnel during plant operation at maximum calculated thermal power and to limit the normal operation radiation levels at the exclusion area boundary to below those levels allowed for continuous nonoccupational exposure. The plant is capable of continued safe operation with fuel cladding defects in the equivalent to one percent of the fuel rods.

In addition, the shielding provided ensures that, in the event of a hypothetical accident, the integrated offsite exposure due to the contained activity does not result in any harmful offsite radiation exposures.

The documents 10 CFR Part 20 and 10 CFR Part 100, or applicable portions thereof, are used as the bases for defining acceptable exposures in areas within and beyond the exclusion area boundary. The access requirements for each piece of plant equipment, instrumentation, or control equipment used in the plant are considered in setting the radiation zone for that area. The major radiation sources considered in the Containment Building are the core fission neutrons and gammas and the N-16 activity while at power and the spent fuel sources during refueling operations. The major source of activity in the Auxiliary Building is that activity associated with the assumption of small cladding defects in the equivalent of one percent of the fuel rods.

The calculation model for determining contained radiation sources used for shielding design is based on the design parameters given in Tables 12.2-12 and 12.2-21 through 12.2-23.

The assumptions used in analyzing postulated accident situations are listed in Table 12.2-24. The sources associated with an equivalent core meltdown accident are consistent with those set forth in TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites [1]. The sources associated with a gap activity release accident are based on the assumption that the fission product activity in the space between the fuel rods and the cladding is released as a result of cladding failure.

Additional assumptions, calculation methods and design bases for contained radiation source data are given in Section 11.1, Source Terms.

12.2.1.6 The Old Steam Generator Storage Facility

The Old Steam Generator Storage Facility is designed to store the four Unit 1 Old Steam Generators and the Old Reactor Vessel Heads and associated Control Rod Drive Mechanisms for Units 1 and 2. The facility satisfies all design requirements and criteria for temporary storage of radioactive materials contained within the Old Steam Generators and Reactor Vessel Heads. The radiological design of the Old Steam Generator storage facility provides adequate shielding to satisfy Comanche Peak licensing basis requirements as per 10CFR Part 20 and incorporates ALARA design features as per Regulatory Guide 8.8.

The maximum contact dose rates at the facility surfaces including walls, door shield blocks, and roofs is less than or equal to 0.25 mR/hr. This results in a radiation zone classification of zone 1 as per UFSAR Table 12.3-6 and allows unrestricted access in the adjacent areas.

Personnel access to the Old Steam Generator Storage Facility openings is controlled by Radiation Protection personnel. Access to the facility will be infrequent, and there are no appreciable normal releases of the radiological materials from the facility.

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

12.2.2.1 Introduction

This section discusses the models, parameters, and assumptions necessary to evaluate airborne concentrations of radionuclides during normal plant operations in areas where personnel access may be required. The plant conditions that are considered for the calculations of expected concentrations of radioactive airborne material include normal power operations, shutdown, and refueling operations. The final design of each reactor unit ensures that the expected airborne isotopic concentrations in the normally accessible areas during normal operating conditions are below the maximum permissible concentrations. Expected airborne concentrations in areas requiring infrequent or periodic access are also acceptable, considering the expected occupancy in the regions.

12.2.2.2 Areas Containing Sources of Airborne Radioactivity

The major buildings considered to contain sources of airborne radioactivity are as follows:

1. Fuel Building
2. Safeguards Buildings
3. Auxiliary Building
4. Turbine Buildings
5. Reactor Containment Buildings

All other areas are considered to have negligible leak sources for airborne radioactivity.

12.2.2.3 Sources of Airborne Radioactive Material

Plant systems and equipment that handle radioactive fluids are potential sources of airborne radioactive material. The most significant of these potential sources which are included in the analysis are the Reactor Coolant System (RCS), Chemical and Volume Control System (CVCS), Residual Heat Removal System (RHRS), Spent Fuel Pools, and the Main Steam System. The specific leakage activities are dependent upon the concentrations of radionuclides in these systems. Potential leakage points considered for these source activities include the opening of normally closed systems and components, such as reactor vessel head removal and servicing of valves and pumps, transfer of spent fuel, and leakage from components in service, such as leakage from valve stem seals and leak-offs, pump shaft seals, relief valve discharges, vents, and drains.

12.2.2.4 Airborne Concentration Calculation Methodology

The basic methodology of NUREG-0017, Revision 1 and conservative assumptions and parameters are used for the calculation of airborne radioactivity concentrations. These assumptions and parameters including the concentrations of radioisotopes in the primary coolant, secondary water and steam are listed in Table 12.2-25. Equations 1 and 2 are used to calculate the airborne concentrations of the radioisotopes presented in Table 12.2-26. For each case, it is assumed that the initial concentration is zero, and that the leakage has occurred long enough to reach the equilibrium condition in that area.

$$C_i(t) = (LR)_i \, A_i \, (PF)_i \, \frac{(1 - e^{-\lambda Ti^t})}{V^{\lambda}Ti} \qquad \text{(Equation 1)}$$

$(LR)_i$ = leak or evaporation rate of the i^{th} radioisotope in the region (grams/sec).

A_i = activity concentration of the i^{th} leaking or evaporating radioisotope in μ Ci/gram.

$(PF)_i$ = partition factor, or fraction of the leaking activity that is airborne for the i^{th} radioisotope.

$^{\lambda}Ti$ = total removal rate constant for the i^{th} isotope from the region (sec^{-1}). This value is the sum of the radioactive decay constant (ln(2) / half life), and the ventilation rate in the room (air changes per second).

t = time interval between the start of the leak and the time at which the concentration is evaluated (sec).

V = volume of the region in which the leak occurs (cc).

Ci(t) = airborne concentration of the ith isotope at time t in the region (μCi/cc).

With high exhaust rates, the equilibrium concentration will be reached within a few hours. From Equation 1, it is evident that this concentration may be determined as shown in Equation 2, below:

$$C_i,Eq = (LR)_i \, A_i \, (PF)_i \, /(V^{\lambda}Ti) \qquad \text{(Equation 2)}$$

The design of the ventilation systems considers airborne radioactivity. This design provides flow paths from areas of low potential airborne concentration to areas of higher potential concentrations. Also, the ventilation design provides exhaust flow directly into the Primary Plant Ventilation Exhaust System supply plenum for most rooms with the highest expected airborne concentrations.

REFERENCES

1. TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites.

2. NUREG-0017, Revision 1, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors.

TABLE 12.2-1
REACTOR COOLANT N-16 ACTIVITY - ORIGINAL LICENSING BASIS[a]

Position in Loop	Loop Transit Time (sec)	N-16 Activity (uCi/g)
Leaving core	0.0	136
Leaving reactor vessel	1.3	113
Entering steam generator	1.7	109
Leaving steam generator	5.8	74
Entering reactor coolant pump	6.5	69
Entering reactor vessel	7.2	65
Entering core	9.2	54
Leaving core	9.9	136

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-2
PRESSURIZER SOURCE ACTIVITIES - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 2)

Isotope	Activity (uCi/g)
Liquid Phase	
N-16 (max.)	1.3
Rb-88	1.1×10^{-2}
Mo-99	2.2
I-131	1.6
I-132	2.0×10^{-2}
I-133	7.0×10^{-1}
I-134	5.5×10^{-3}
I-135	1.4×10^{-1}
Cs-134	2.5×10^{-1}
Cs-136	1.1×10^{-1}
Cs-137	1.3
Cs-138	5.5×10^{-3}
Steam Phase	
Kr-85	5.1×10^{1}
Kr-85M	1.0×10^{-1}
Kr-87	1.8×10^{-2}
Kr-88	1.2×10^{-1}
Xe-131M	4.7
Xe-133	3.6×10^{2}

TABLE 12.2-2
PRESSURIZER SOURCE ACTIVITIES - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 2)

Isotope	Activity (uCi/g)
Xe-133M	1.8
Xe-135	6.5×10^{-1}
Xe-135M	5.0×10^{-4}
Xe-138	2.2×10^{-3}
Deposited Crud	
Cr-51	9.8×10^{-2}
Mn-54	1.5×10^{-1}
Mn-56	2.2×10^{-2}
Co-58	3.8
Co-60	1.6×10^{-1}
Fe-59	1.4×10^{-1}

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-3
DESIGN BASIS REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITY - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 2)

Isotope	Activity (uCi/g)	Isotope	Activity (uCi/g)
H-3	3.5 (max.)	Cs-136	0.15
Br-84	4.3×10^{-2}	Cs-137	1.5
Rb-88	3.7	Cs-138	0.98
Rb-89	1.1×10^{-1}	Ba-140	4.3×10^{-3}
Sr-89	3.3×10^{-3}	La-140	1.5×10^{-3}
Sr-90	1.7×10^{-4}	Ce-144	3.4×10^{-4}
Sr-91	1.9×10^{-3}	Pr-144	3.4×10^{-4}
Sr-92	7.4×10^{-4}	Kr-85	8.8 (peak)
Y-90	2.0×10^{-4}	Kr-85M	2.1
Y-91	6.1×10^{-3}	Kr-87	1.2
Y-92	7.2×10^{-4}	Kr-88	3.7
Zr-95	7.0×10^{-4}	Xe-131M	1.9
Nb-95	6.9×10^{-4}	Xe-133	2.81×10^{2}
Mo-99	5.3	Xe-133M	3.1
I-131	2.5	Xe-135	6.3
I-132	0.9	Xe-135M	0.7
I-133	4.0	Xe-138	0.7
I-134	0.6	Cr-51	9.5×10^{-4}
I-135	2.2	Mn-54	7.9×10^{-4}
Te-132	0.26	Mn-56	3.0×10^{-2}
Te-134	2.9×10^{-2}	Co-58	2.6×10^{-2}
Cs-134	0.3	Co-60	7.7×10^{-4}

TABLE 12.2-3
DESIGN BASIS REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITY - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 2)

Fe-59	1.1×10^{-3}

a) Historical. Not subject to future updating. Has been retained to preserve original design basis. Design basis RCS source term for SPU conditions presented in Table 12.2-3A.

TABLE 12.2-4
NEUTRON AND GAMMA FLUXES[a] AT THE SURFACE OF THE PRIMARY CONCRETE - ORIGINAL LICENSING BASIS[b]

(Sheet 1 of 2)

Maximum Neutron Fluxes Incident on the Primary Concrete

E > 1.0 MeV	7.6×10^{8} n/cm^2-sec
5.53 KeV < E < 1.0 MeV	1.2×10^{10} n/cm^2-sec
0.625 eV E < 5.53 keV	7.1×10^{9} n/cm^2-sec
E < 0.625 eV	1.8×10^{9} n/cm^2-sec

Maximum Gamma Ray Energy Fluxes Incident on the Primary Concrete

Group	Flux (MeV/cm^2-sec)	Group Energy (MeV)
1	3.7×10^{9}	7.5
2	3.3×10^{9}	4.0
3	1.7×10^{9}	2.5
4	1.0×10^{9}	0.8

Normalized Axial Variation for Neutron and Gamma Fluxes

Distance Above and Below Core Midplane (ft)	F
0	1.0
1	1.0
2	0.99
3	0.93
4	0.76
5	0.51
6	0.30

TABLE 12.2-4

NEUTRON AND GAMMA FLUXES[a] AT THE SURFACE OF THE PRIMARY CONCRETE - ORIGINAL LICENSING BASIS[b]

(Sheet 2 of 2)

a) Fluxes given at core midplane

b) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-5
RADIATION SOURCE ACITVITIES FOR EQUIPMENT IN THE CHEMICAL AND VOLUME CONTROL SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 3)

Letdown Coolant Fission and Corrosion Product Activity

Gamma Energy (MeV)	Specific Activity (MeV/g-sec)
0.4	4.5×10^5
0.8	2.7×10^5
1.3	1.7×10^5
1.7	1.2×10^5
2.2	1.4×10^5
2.5	1.6×10^5
3.5	1.9×10^4

Filter Activities

Gamma Energy (MeV)	Reactor Coolant (MeV/cm^3-sec)	Seal Water Injection (MeV/cm^3-sec)	Seal Water Return (MeV/cm^3-sec)
0.8	5.7×10^7	4.8×10^7	1.1×10^7
1.3	1.5×10^7	1.2×10^7	3.0×10^6

Demineralizer Activities

Gamma Energy (MeV)	Mixed Bed (MeV/cm^3-sec)	Cation Bed (MeV/cm^3-sec)	Boron Thermal Regeneration[b] (MeV/cm^3-sec)
0.4	1.5×10^8	1.9×10^6	2.5×10^6
0.8	3.4×10^8	3.3×10^8	1.7×10^6
1.3	3.3×10^7	4.8×10^6	6.8×10^5
1.7	1.7×10^7	3.4×10^6	3.3×10^5
2.2	4.5×10^6	-	1.2×10^5
2.5	1.0×10^5	-	-

TABLE 12.2-5
RADIATION SOURCE ACITVITIES FOR EQUIPMENT IN THE CHEMICAL AND VOLUME CONTROL SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 3)

3.5	2.0×10^5	-	-

Volume Control Tank

Gamma Energy (MeV)	Vapor Phase (MeV/cm^3-sec)	Liquid Phase[b] (MeV/gm-sec)
0.1	2.3×10^6	3.1×10^5
0.4	5.0×10^5	7.0×10^4
0.8	1.7×10^5	4.5×10^4
1.3	-	1.6×10^4
1.7	1.3×10^5	2.6×10^4
2.2	3.0×10^5	4.0×10^4
2.5	6.2×10^5	8.3×10^4
3.5	-	1.7×10^3

Regenerative Heat Exchanger and Excess Letdown Heat Exchanger (Specific Source Strength)

Gamma Energy (MeV)	Regenerative-Shell Excess Letdown Tubes (MeV/g-sec)	Regenerative[c] Tubes (MeV/g-sec)
0.4	4.5×10^5	3.8×10^5
0.8	2.7×10^5	4.5×10^4
1.3	1.7×10^5	1.6×10^4
1.7	1.2×10^5	2.6×10^4
2.2	1.4×10^5	4.0×10^4
2.5	1.6×10^5	8.3×10^4
3.5	1.9×10^4	1.7×10^3
6.1	2.2×10^6	-
7.1	1.8×10^5	-

TABLE 12.2-5
RADIATION SOURCE ACITVITIES FOR EQUIPMENT IN THE CHEMICAL AND VOLUME CONTROL SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 3)

Thermal Regeneration

	Letdown Heat Exchanger, Tube Side Seal Water Heat Exchanger, Tube Side	Moderating Heat Exchanger, Tube and Shell Sides, Chiller Heat Exchanger, Tube Side, Letdown Reheat Heat Exchanger Tube and Shell Sides
Gamma Energy (MeV)	Specific Source Strength (MeV/g-sec)	Specific Source Strength (MeV/g-sec)
0.4	4.5×10^5	4.2×10^5
0.8	2.7×10^5	7.6×10^5
1.3	1.7×10^5	1.6×10^4
1.7	1.2×10^5	3.9×10^4
2.2	1.4×10^5	1.2×10^5
2.5	1.6×10^5	1.6×10^5
3.5	1.9×10^4	1.7×10^3

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

b) These data are also applicable to charging pump sources.

c) Modified to account for activity removed by demineralizers and volume control tank

TABLE 12.2-6
HEAT EXCHANGER DESIGN DATA

(Sheet 1 of 2)

Heat Exchanger		Fluid Volume (ft^3)	
		Shell	Tubes
1.	Excess letdown	4.06	1.15
2.	Letdown	195 gal	80 gal
3.	Letdown chiller	23.08	9.50
4.	Letdown reheat	1.12	0.85
5.	Moderating	17.20	9.80
6.	Reactor coolant drain tank	6.72	3.25
7.	Regenerative	7.00	1.33
8.	Residual	174.00	112.00
9.	Seal water	16.86	7.89

	Length (ft)	OD (in.)	Shell Thickness (in.)
1.	12.578	8.625	0.322
2.	16.260	22.00	0.375
3.	15.167	20.00	0.375
4.	5.667	8.625	0.148
5.	16.292	18.00	0.1875
6.	13.760	12.75	0.330
7.	-	8.625	0.75
8.	22.583	44.25	0.50
9.	11.917	20.00	0.375

TABLE 12.2-6
HEAT EXCHANGER DESIGN DATA

(Sheet 2 of 2)

	Tubes			
	Length (ft)	Number ID	ID (in.)	OD (in.)
1.	23.25	25 U's	0.495	0.625
2.	27.9	142 U's	0.652	0.75
3.	26.25	94 U's	0.652	0.75
4.	8.9	15 U's	0.652	0.75
5.	29	101 U's	0.652	0.75
6.	24.75	40 U's	0.652	0.75
7.	33	108	0.245	0.375
8.	46.5	729 U's	0.652	0.75
9.	19.6	94 U's	0.652	0.75

TABLE 12.2-7
RADIATION SOURCE ACTIVITIES FOR EQUIPMENT IN THE BORON RECYCLE SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 3)

Filter Activities

Gamma Energy (MeV)	Recycle Evaporator Feed (MeV/cm^3-sec)	Recycle Evaporator Condensate (MeV/cm^3-sec)	Recycle Evaporator Concentrates (MeV/cm^3-sec)
0.4	-	1.6×10^5	-
0.8	1.1×10^7	8.0×10^4	1.2×10^4
1.3	3.0×10^6	3.3×10^4	-

Demineralizer Activities

Gamma Energy (MeV)	Recycle Evaporator Feed (MeV/cm^3-sec)	Evaporator Condensate (MeV/cm^3-sec)
0.4	1.4×10^7	4.4×10^4
0.8	4.5×10^7	2.2×10^4
1.3	2.2×10^6	5.5×10^3
1.7	1.4×10^6	2.6×10^3
2.2	3.3×10^5	1.0×10^3

TABLE 12.2-7
RADIATION SOURCE ACTIVITIES FOR EQUIPMENT IN THE BORON RECYCLE SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 3)

Recycle Evaporator and Waste Evaporator

Gamma Energy (MeV)	Vent Condenser Vapor Source Strength (MeV/cm3-sec)	Evaporator Concentrates Source Strength (MeV/gm-sec)
0.1	1.1×10^{7}	-
0.4	3.6×10^{6}	-
0.8	1.5×10^{6}	1.2×10^{6}
1.3	-	-
1.7	1.0×10^{6}	-
2.2	3.8×10^{6}	-
2.5	5.2×10^{6}	-

Recycle Holdup Tanks

Gamma Energy (MeV)	Vapor Phase (MeV/cm3-sec)	Liquid Phase[b] (MeV/g-sec)
0.1	2.1×10^{6}	3.1×10^{5}
0.4	4.4×10^{5}	1.1×10^{5}
0.8	1.2×10^{5}	4.9×10^{4}
1.3	-	1.7×10^{3}
1.7	9.3×10^{4}	3.1×10^{4}
2.2	1.9×10^{5}	1.1×10^{5}
2.5	4.3×10^{5}	1.6×10^{5}

TABLE 12.2-7
RADIATION SOURCE ACTIVITIES FOR EQUIPMENT IN THE BORON RECYCLE SYSTEM - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 3)

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

b) These data are also applicable to the recycle evaporator feed pump.

TABLE 12.2-8
RADIATION SOURCE ACTIVITIES FOR EQUIPMENT IN THE STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM - ORIGINAL DESIGN BASIS[a]

Filter Activities

Gamma Energy (MeV)	Inlet (MeV/cm3-sec)	Resin Sluice (MeV/cm3-sec)
0.8	4.02×10^{5}	1.24×10^{4}
1.3	1.10×10^{5}	4.45×10^{2}
1.7	8.07×10^{3}	6.05×10^{1}
2.5	1.62×10^{2}	9.68×10^{-1}
4.0	5.58	4.34×10^{-2}
5.0	-	7.29×10^{-3}

Demineralizer and Spent Resin Storage Tank Activities

Gamma Energy (MeV)	Demineralizers (MeV/cm3-sec)	Spent Resin Storage Tank (MeV/cm3-sec)
0.8	1.24×10^{5}	7.22×10^{4}
1.3	4.45×10^{3}	2.60×10^{3}
1.7	6.05×10^{2}	3.54×10^{2}
2.5	9.68	5.66
4.0	4.34×10^{-1}	2.53×10^{-1}
5.0	7.29×10^{-2}	4.26×10^{-2}

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-9
RADIATION SOURCE ACTIVITY FOR THE RESIDUAL HEAT REMOVAL LOOP - ORIGINAL DESIGN BASIS[a]

Gamma Energy (MeV)	Specific Source Strength (MeV/g-sec)
0.4	3.4×10^5
0.8	1.3×10^5
1.3	4.4×10^4
1.7	2.9×10^4
2.2	2.5×10^4
2.5	4.1×10^4

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-10
RADIATION SOURCE ACTIVITY FOR EQUIPMENT IN THE SPENT FUEL POOL COOLING AND PURIFICATION SYSTEM - ORIGINAL LICENSING BASIS[a]

Gamma Energy (MeV)	Demineralizer (MeV/cm^3-sec)	Filter (MeV/cm^3-sec)
0.4	2.1×10^6	-
0.8	7.4×10^5	1.1×10^7
1.3	4.1×10^3	3.0×10^6
1.7	4.7×10^3	-

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-11
CORE SHUTDOWN SOURCES - ORIGINAL LICENSING BASIS[a]

Core Shutdown Sources (MeV/cm^3-sec)

Photon Energy (MeV)	Time After Shutdown		
	4 Hours	12 Hours	1 Day
0.4	3.1×10^{11}	2.3×10^{11}	1.9×10^{11}
0.8	1.3×10^{12}	9.8×10^{11}	8.0×10^{11}
1.3	3.9×10^{11}	2.9×10^{11}	2.5×10^{11}
1.7	5.1×10^{11}	3.8×10^{11}	3.3×10^{11}
2.2	7.2×10^{10}	2.6×10^{10}	1.5×10^{10}
2.5	8.9×10^{10}	4.7×10^{10}	3.7×10^{10}
3.5	8.2×10^{9}	2.0×10^{9}	1.3×10^{9}

Photon Energy (MeV)	Time After Shutdown		
	1 Week	1 Month	3 Months
0.4	9.2×10^{10}	3.8×10^{10}	1.3×10^{10}
0.8	4.0×10^{11}	2.3×10^{11}	1.2×10^{11}
1.3	1.6×10^{11}	1.2×10^{11}	5.8×10^{10}
1.7	2.3×10^{11}	6.2×10^{10}	2.9×10^{9}
2.2	8.5×10^{9}	6.7×10^{9}	5.0×10^{9}
2.5	2.5×10^{10}	7.9×10^{9}	3.5×10^{8}
3.5	9.6×10^{8}	2.0×10^{8}	1.5×10^{7}

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-12
REACTOR CORE COMPOSITION - ORIGINAL LICENSING BASIS[a]

Material	Volume Fractions
UO_2	0.3052
Zirconium	0.0943
Stainless steel	0.0053
Inconel	0.0043
Water	0.5909

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-13
CONTROL ROD SOURCES - ORIGINAL LICENSING BASIS[a]

Irradiated Ag-In-Cd Control Rod Sources (MeV/cm-sec/rod)

Photon Energy (MeV)	Time After Shutdown	
	0	1 Week
0.8	3.2×10^{12}	3.2×10^{12}
1.3	9.0×10^{11}	8.9×10^{11}

Photon Energy (MeV)	Time After Shutdown		
	1 Month	6 Months	1 Year
0.8	3.0×10^{12}	2.0×10^{12}	1.3×10^{12}
1.3	8.4×10^{11}	5.8×10^{11}	3.8×10^{11}

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-14
INCORE DETECTOR SOURCES - ORIGINAL LICENSING BASIS[a]

Irradiated In-Core Detector Drive Wire Sources (MeV/cm-sec)

Photon Energy (MeV)	Time After Shutdown		
	0	4 Hours	12 Hours
0.8	4.1×10^{10}	1.1×10^{10}	2.6×10^{9}
1.3	1.2×10^{9}	1.2×10^{9}	1.2×10^{9}
2.2	4.0×10^{10}	1.3×10^{10}	1.5×10^{9}

Photon Energy MeV)	Time After Shutdown		
	1 Day	1 Week	1 Month
0.8	1.1×10^{9}	1.0×10^{9}	1.0×10^{9}
1.3	1.2×10^{9}	1.1×10^{9}	8.4×10^{8}
2.2	5.8×10^{7}	-	-

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-15
DEPOSITED CORROSION PRODUCT ACTIVITY ON STEAM GENERATOR SURFACES - ORIGINAL LICENSING BASIS[a]

Deposited Activity (uCi/cm^2)

	Operating Time (Months)				
Isotope	0	6	12	24	36
Mn-54	0.0	0.15	0.60	1.5	2.0
Mn-56	0.0	3.3	3.3	3.3	3.3
Co-58	0.0	4.5	10.2	11.0	11.0
Fe-59	0.0	1.4	3.0	3.0	3.0
Co-60	0.0	0.20	0.80	2.0	3.5

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-16
POST-LOCA RADIATION SOURCES

Radiation Source	Fraction of Core Activity Release
Primary coolant (Pressurized LOCA)	100% noble gases, 50% halogens, 1% remainder
Sump Water (Depressurized LOCA)	50% halogens, 1% remainder
Containment atmosphere (Inside containment)	100% noble gases, 50% halogens

TABLE 12.2-17
SOURCE STRENGTH FOR THE POST-ACCIDENT UNIT 1 CONTAINMENT ATMOSPHERE[a] - ORIGINAL LICENSING BASIS[b]

Source Strength at Time After Accident (MeV/cc-sec)

MeV/Gamma	0 HR	0.5 HR	1 HR	2 HR	8 HR
0.4	6.41E+07	2.24E+07	1.98E+07	1.81E+07	1.56E+07
0.8	2.40E+08	1.54E+08	1.25E+08	8.81E+07	3.16E+07
1.3	1.43E+08	5.88E+07	5.34E+07	4.16E+07	1.81E+07
1.7	1.39E+08	5.06E+07	4.01E+07	2.15E+07	1.13E.07
2.2	6.37E+07	2.64E+07	1.81E+07	1.23E+07	2.95E+06
2.5	6.71E+07	3.77E+07	3.15E+07	2.36E+07	5.31E+06
3.5	6.43E+07	1.63E+06	1.12E+06	4.47E+05	2.53E+04
6.15	8.67E+06	1.38E+04	6.76E+03	1.79E+03	0.00E+00

MeV/Gamma	24 HR	168 HR	720 HR	4380 HR	8760 HR
0.4	1.18E+07	5.06E+06	5.69E+05	1.27E+00	0.00E+00
0.8	1.48E+07	1.27E+06	1.31E+05	6.33E+02	5.90E+02
1.3	4.06E+06	6.33E+04	3.92E+02	0.00E+00	0.00E+00
1.7	1.96E+06	2.99E+04	2.11E+02	0.00E+00	0.00E+00
2.2	1.81E+05	7.17E+03	5.48E+01	0.00E+00	0.00E+00
2.5	2.07E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3.5	3.58E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
6.15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Note:

a. Based on minimum total dry containment volume of 2.985×10^6 ft^3 and release of 100 percent of core noble gases and 50 percent of core halogens to the containment atmosphere.

b. Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-17A
SOURCE STRENGTH FOR THE POST-ACCIDENT UNIT 2 CONTAINMENT ATMOSPHERE[a][b] - ORIGINAL LICENSING BASIS[c]

Source Strength at Time After Accident (MeV/cc-sec)

Mev/Gamma	0 HR	0.5 HR	1 HR	2 HR	8 HR
0.4	6.32E+07	2.20E+07	1.95E+07	1.79E+07	1.54E+07
0.9	2.37E+08	1.52E+08	1.23E+08	8.68E+07	3.11E+07
1.35	1.41E+08	5.79E+07	5.26E+07	4.10E+07	1.79E+07
1.8	1.37E+08	4.99E+07	3.95E+07	2.47E+07	1.12E+07
2.2	6.27E+07	2.60E+07	1.78E+07	1.21E+07	2.91E+06
2.6	6.61E+07	3.71E+07	3.10E+07	2.33E+07	5.24E+06
3.0	3.23E+07	4.07E+05	2.83E+05	1.64E+05	2.08E+04
4.0	2.16E+07	1.43E+06	8.10E+05	2.74E+05	4.10E+03
5.0	1.71E+07	2.72E+04	1.33E+04	3.53E+03	0.0
0.0	0.0	0.0	0.0	0.0	0.0

Mev/Gamma	24 HR	168 HR	720 HR	4380 HR	8760 HR
0.4	1.16E+07	4.99E+06	5.61E+05	1.25E+00	0.0
0.9	1.46E+07	1.25E+06	1.29E+05	6.23E+02	5.82E+02
1.35	4.00E+06	6.23E+04	3.86E+02	0.0	0.0
1.8	1.93E+06	2.95E+04	2.08E+02	0.0	0.0
2.2	1.78E+05	7.06E+03	5.41E+01	0.0	0.0
2.6	2.04E+05	0.0	0.0	0.0	0.0
3.0	3.53E+02	0.0	0.0	0.0	0.0
4.0	0.0	0.0	0.0	0.0	0.0
5.0	0.0	0.0	0.0	0.0	0.0
6.0	0.0	0.0	0.0	0.0	0.0

Note:

a. Based on a minimum total dry containment volume of 3.03E06 cu.ft. and release of 100 percent of core noble gases and 50 percent of core halogens to the containment atmosphere.

b. Source strength data from Westinghouse Radiation Analysis Manual and normalized to approximate power level and reactor containment building volume.

c. Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-18
DELETED

TABLE 12.2-18A
DELETED

TABLE 12.2-19
SOURCE STRENGTH FOR THE POST-ACCIDENT UNIT 1 PRESSURIZED PRIMARY COOLANT[a][b] - ORIGINAL LICENSING BASIS[c]

Source Strength at Time After Accident (MeV/cc-sec)

MeV/Gamma	0 HR	0.5 HR	1 HR	2 HR	8 HR
0.4	1.85E+10	6.53E+09	5.73E+09	5.18E+09	4.44E+09
0.8	6.91E+10	4.58E+10	3.74E+10	2.64E+10	9.80E+10
1.3	4.15E+10	1.95E+10	1.66E+10	1.21E+10	5.11E+09
1.7	4.02E+10	2.01E+10	1.58E+10	8.88E+09	3.56E+09
2.2	1.180+10	1.01E+10	7.89E+09	5.94E+09	1.39E+09
2.5	1.89E+10	1.26E+10	1.02E+10	7.07E+09	1.49E+09
3.5	1.86E+09	2.32E+09	1.82E+09	9.98E+08	1.45E+08
6.15	2.49E+09	1.92E+07	1.87E+07	1.50E+07	3.37E+06

MeV/Gamma	24 HR	168 HR	720 HR	4380 HR	8760 HR
0.4	3.38E+09	1.44E+09	1.71E+08	3.02E+06	1.74E+06
0.8	4.83E+09	8.83E+08	4.20E+08	1.05E+08	3.74E+07
1.3	1.19E+09	5.53E+07	1.41E+07	2.09E+06	1.51E+06
1.7	8.90E+08	2.64E+08	7.68E+07	1.28E+06	9.41E+05
2.2	7.31E+07	1.06E+07	4.32E+06	1.39E+06	8.72E+05
2.5	7.79E+07	1.63E+07	4.65E+06	1.39E+03	0.00E+00
3.5	3.14E+06	3.88E+05	1.12E+05	0.00E+00	0.00E+00
6.15	6.39E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Note:

a. Based on the total coolant volume of 10833 ft^3 which is equal to total RCS liquid volume of 11500 ft^3 at 590°F and 2250 psia minus pressurizer liquid volume of 1080 ft^3 plus RHR Train A total volume of 413 ft^3.

b. 100% Noble gases, 50% halogens, and 1% remainder.

c. Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-19A
SOURCE STRENGTH FOR THE UNIT 1 SUMP WATER[a][b] - ORIGINAL LICENSING BASIS[c]

Source Strength at Time After Accident (MeV/cc-sec)

MeV/Gamma	0 HR	0.5 HR	1 HR	2 HR	8 HR
0.4	7.97E+08	5.34E+08	4.75E+08	4.20E+08	3.71E+08
0.8	9.07E+09	7.34E+09	6.04E+09	4.24E+09	1.60E+09
1.3	4.63E+09	3.22E+09	2.77E+09	2.04E+09	8.77E+08
1.7	3.38E+09	2.83E+09	2.34E+09	1.33E+09	5.65E+08
2.2	3.00E+08	6.63E+08	6.30E+08	5.53E+08	1.39E+08
2.5	6.65E+08	4.75E+08	3.41E+08	1.59E+08	3.77E+07
3.5	1.70E+09	3.93E+08	3.00E+08	1.63E+08	2.42E+07
6.15	4.05E+08	3.32E+06	3.28E+06	2.64E+06	5.91E+03

MeV/Gamma	24 HR	168 HR	720 HR	4380 HR	8760 HR
0.4	3.28E+08	1.91E+08	2.69E+07	5.30E+05	3.06E+05
0.8	8.26E+08	1.55E+08	7.36E+07	1.84E+07	6.53E+06
1.3	2.08E+08	9.18E+06	2.47E+06	3.67E+05	2.65E+05
1.7	1.55E+08	4.63E+07	1.35E+07	2.24E+05	1.65E+05
2.2	1.08E+07	1.86E+06	7.57E+05	2.45E+05	1.53E+05
2.5	9.58E+06	2.85E+06	8.16E+05	2.45E+02	0.00E+00
3.5	5.33E+05	6.81E+04	1.96E+04	0.00E+00	0.00E+00
6.15	1.12E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Note:

a. Based on the total sump water volume of 61740 ft^3.

b. 0% noble gases, 50% halogens, 1% remainder.

c. Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-19B

SOURCE STRENGTH FOR THE POST-ACCIDENT UNIT 1 PRIMARY COOLANT SAMPLE FLUID[a] - ORIGINAL LICENSING BASIS[b]

Source Strength at Time After Accident (MeV/cc-sec)

MeV/Gamma	0 HR	0.5 HR	1 HR	2 HR	8 HR
0.4	2.61E+10	9.22E+09	8.09E+09	7.32E+09	6.27E+09
0.8	9.76E+10	6.46E+10	5.28E+10	3.72E+10	1.38E+10
1.3	5.86E+10	2.75E+10	2.34E+10	1.71E+10	7.22E+09
1.7	5.68E+10	2.84E+10	2.23E+10	1.25E+10	5.02E+09
2.2	2.54E+10	1.42E+10	1.11E+10	8.38E+09	1.97E+09
2.5	2.67E+10	1.78E+10	1.44E+10	9.97E+08	2.11E+09
3.5	2.62E+10	3.28E+09	2.57E+09	1.41E+09	2.05E+08
6.15	3.51E+09	2.71E+07	2.64E+07	2.12E+07	4.76E+06

MeV/Gamma	24 HR	168 HR	720 HR	4380 HR	8760 HR
0.4	4.77E+09	2.03E+09	2.41E+08	4.27E+06	2.46E+06
0.8	6.82E+09	1.25E+09	5.92E+08	1.48E+08	5.27E+07
1.3	1.67E+09	7.38E+07	1.98E+07	2.95E+06	2.13E+06
1.7	1.26E+09	3.73E+08	1.08E+08	1.80E+06	1.33E+06
2.2	1.03E+08	1.49E+07	6.09E+06	1.97E+06	1.23E+06
2.5	1.10E+08	2.30E+07	6.56E+06	1.97E+03	0.00E+00
3.5	4.43E+06	5.48E+05	1.57E+05	0.00E+00	0.00E+00
6.15	9.02E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Note:

a. Based on source strengths for the primary coolant given in Table 12.2-19 which are divided by a density correction factor of 0.7084 to obtain the sample water source strengths.

b. Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-19C
SOURCE STRENGTH FOR THE POST-ACCIDENT UNIT 2 PRESSURIZED PRIMARY COOLANT[a][b][c] - ORIGINAL LICENSING BASIS[d]

Source Strength at Time After Accident (MeV/cc-sec)

Mev/Gamma	0 HR	0.5 HR	1 HR	2 HR	8 HR
0.4	1.85E+10	6.53E+09	5.73E+09	5.18E+09	4.44E+09
0.9	6.91E+10	4.58E+10	3.74E+10	2.64E+10	9.80E+09
1.35	4.15E+10	1.95E+10	1.66E+10	1.21E+10	5.11E+09
1.8	4.02E+10	2.01E+10	1.58E+10	8.88E+09	3.56E+09
2.2	1.80E+10	1.01E+10	8.01E+09	5.94E+09	1.39E+09
2.6	1.89E+10	1.26E+10	1.02E+10	7.06E+09	1.49E+09
3.0	9.47E+10	1.51E+09	1.24E+09	6.97E+08	9.99E+07
4.0	6.65E+09	8.63E+07	5.64E+08	2.86E+08	4.18E+07
5.0	4.93E+09	3.78E+07	3.74E+07	3.00E+07	6.74E+06
6.0	2.21E+07	2.79E+05	5.81E+02	0.0	0.0

Mev/Gamma	24 HR	168 HR	720 HR	4380 HR	8760 HR
0.4	3.38E+09	1.44E+09	1.71E+08	3.02E+06	1.74E+06
0.9	4.83E+09	8.83E+08	4.20E+08	1.05E+08	3.73E+07
1.35	1.19E+09	5.23E+07	1.41E+07	2.09E+06	1.51E+06
1.8	8.90E+08	2.64E+08	7.67E+07	1.28E+06	9.41E+05
2.2	7.31E+07	1.06E+07	4.31E+06	1.39E+06	8.71E+05
2.6	7.79E+07	1.63E+07	4.65E+06	1.39E+03	0.0
3.0	2.19E+06	2.79E+05	8.02E+04	0.0	0.0
4.0	8.83E+05	1.09E+05	3.14E+04	0.0	0.0
5.0	1.28E+05	0.0	0.0	0.0	0.0
6.0	0.0	0.0	0.0	0.0	0.0

Note:

a. Based on the total coolant volume of 10833 cu.ft. which is equal to total RCS liquid volume of 11500 cu.ft. at 590ºF and 2250 psia minus pressurizer volume of 1080 cu.ft. plus RHR train A total volume of 413 cu.ft.

b. 100% noble gases, 50% halogens and 1% remainder.

c. Source strength data from Westinghouse Radiation Analysis Manual and normalized to approximate power level and total coolant volume.

d. Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-19D
SOURCE STRENGTH FOR THE UNIT 2 SUMP WATER[a][b][c] - ORIGINAL LICENSING BASIS[d]

Source Strength at Time After Accident (MeV/cc-sec)

Mev/Gamma	0 HR	0.5 HR	1 HR	2 HR	8 HR
0.4	7.71E+08	5.16E+08	5.49E+08	4.06E+08	3.59E+08
0.9	8.77E+09	7.10E+09	5.84E+09	4.10E+09	1.55E+09
1.35	4.47E+09	3.11E+09	2.68E+09	1.97E+09	8.48E+08
1.8	3.27E+09	2.74E+09	2.27E+09	1.48E+09	5.46E+08
2.2	2.90E+08	6.41E+08	6.49E+08	5.34E+08	1.34E+08
2.6	6.43E+08	4.59E+08	3.29E+08	1.54E+08	3.65E+07
3.0	8.18E+08	2.43E+08	2.00E+08	1.11E+08	1.60E+07
4.0	4.38E+08	1.34E+08	8.67E+07	4.34E+07	6.90E+06
5.0	7.75E+08	6.33E+06	6.35E+06	5.10E+06	1.14E+06
6.0	3.75E+06	4.73E+04	9.86E+01	0.0	0.0

Mev/Gamma	24 HR	168 HR	720 HR	4380 HR	8760 HR
0.4	3.17E+08	1.85E+08	2.60E+07	5.13E+05	2.96E+05
0.9	7.98E+08	1.50E+08	7.12E+07	1.77E+07	6.31E+06
1.35	2.01E+08	8.87E+06	2.38E+06	3.55E+05	2.56E+05
1.8	1.50E+08	4.48E+07	1.30E+07	2.17E+05	1.60E+05
2.2	1.04E+07	1.79E+06	7.32E+05	2.37E+05	1.48E+05
2.6	9.27E+06	2.76E+06	7.89E+05	2.37E+02	0.0
3.0	3.55E+05	4.73E+04	1.36E+04	0.0	0.0
4.0	1.50E+05	1.85E+04	5.32E+03	0.0	0.0
5.0	2.17E+04	0.0	0.0	0.0	0.0
6.0	0.0	0.0	0.0	0.0	0.0

Note:

a. Based on total sump water volume from: 388,677 gal from Refueling Water Storage Tank, 3800 cu.ft. from accumulators, and 11500 cu.ft. (@ 590°F) of Reactor Coolant.

b. 0% noble gases, 50% halogens and 1% remainder.

TABLE 12.2-19D
SOURCE STRENGTH FOR THE UNIT 2 SUMP WATER[a][b][c] -
ORIGINAL LICENSING BASIS[d]

c. Source strength data from Westinghouse Radiation Analysis Manual and normalized to approximate power level and total coolant volume.

d. Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-19E
SOURCE STRENGTH FOR THE POST-ACCIDENT UNIT 2 PRIMARY COOLANT SAMPLE FLUID[a][b] - ORIGINAL LICENSING BASIS[c]

Source Strength at Time After Accident (MeV/cc-sec)

Mev/Gamma	0 HR	0.5 HR	1 HR	2 HR	8 HR
0.4	2.61E+10	9.22E+09	8.09E+09	7.31E+09	6.27E+09
0.9	9.75E+10	6.47E+10	5.28E+10	3.73E+10	1.38E+10
1.35	5.86E+10	2.75E+10	2.34E+10	1.71E+10	7.21E+09
1.8	5.67E+10	2.84E+10	2.23E+10	1.25E+10	5.03E+09
2.2	2.54E+10	1.43E+10	1.13E+10	8.39E+09	1.96E+09
2.6	2.67E+10	1.78E+10	1.44E+10	9.97E+09	2.10E+09
3.0	1.34E+10	2.13E+09	1.75E+09	9.84E+08	1.41E+08
4.0	9.39E+09	1.22E+09	7.96E+08	4.04E+08	5.90E+07
5.0	6.96E+09	5.34E+07	5.28E+07	4.23E+07	9.51E+06
6.0	3.12E+07	3.94E+05	8.20E+02	0.0	0.0

Mev/Gamma	24 HR	168 HR	720 HR	4380 HR	8760 HR
0.4	4.77E+09	2.03E+09	2.41E+08	4.26E+06	2.46E+06
0.9	6.82E+09	1.25E+09	5.93E+08	1.48E+08	5.27E+07
1.35	1.68E+09	7.38E+07	1.99E+07	2.95E+06	2.13E+06
1.8	1.26E+09	3.73E+08	1.08E+08	1.81E+06	1.33E+06
2.2	1.03E+08	1.50E+07	6.08E+06	1.96E+06	1.23E+06
2.6	1.10E+08	2.30E+07	6.56E+06	1.96E+03	0.0
3.0	3.09E+06	3.94E+05	1.13E+05	0.0	0.0
4.0	1.25E+06	1.54E+05	4.43E+04	0.0	0.0
5.0	1.81E+05	0.0	0.0	0.0	0.0
6.0	0.0	0.0	0.0	0.0	0.0

Note:

a. Based on source strengths for the primary coolant given in Table 12.2-19C which are divided by the density correction factor of 0.7084 to obtain the sample water source strengths.

b. Source strength data from Westinghouse Radiation Analysis Manual and normalized to approximate power level and total coolant volume.

c. Historical. Not subject to future updating. Has been retained to preserve original design basis.

TABLE 12.2-20
RADIOACTIVE SOURCES

(Sheet 1 of 2)

	Byproduct, source, and/or special nuclear material	Form		Maximum quantity	
A.	Any radioactive material with Atomic Numbers 1 through 83, inclusive	A.	Any	A.	Not to exceed 1 curie per radionuclide, except: Hydrogen-3 6 curies Sodium-24 3 curies
B.	Any radioactive material with Atomic Numbers 84, 85, 87, 89, 91, 93, 95, or 96	B.	Any	B.	Not to exceed 500 microcuries per radionuclide
C.	Any radioactive material with Atomic Numbers 4 through 84, inclusive	C.	Sealed or plated sourced or foils	C.	Not to exceed 10 curies per source or foil, except: Cesium-137 6000 curies
D.	Any radioactive material with Atomic Number 85, 87, 89, 91, 93, 95, or 96	D.	Sealed or plated sources or foils	D.	Not to exceed 2 millicuries per source or foil, except: Americium-241 4 curies
E.	Thorium-230	E.	Any	E.	.9 gram
F.	Uranium-233	F.	Any	F.	.9 gram
G.	Uranium-234	G.	Any	G.	.9 gram
H.	Uranium-235	H.	Any	H.	.9 gram
I.	Uranium-238	I.	Any	I.	.9 gram

TABLE 12.2-20
RADIOACTIVE SOURCES

(Sheet 2 of 2)

	Byproduct, source, and/or special nuclear material	Form		Maximum quantity	
J.	Plutonium-238	J.	Any	J.	.9 gram
K.	Plutonium-239	K.	Any	K.	.9 gram
L.	Plutonium-238	L.	Sealed sources	L.	One source of 2.2 curies
M.	Californium-252	M.	Sealed sources	M.	600 millicuries
N.	Uranium enriched in the U-235 isotope	N.	In unirradiated reactor fuel assemblies	N.	See Technical Specification 4.3, “Fuel Storage”
O.	Uranium enriched in the U-235 isotope	O.	Contained in excore neutron detectors	O.	40 grams contained U-235 in uranium enriched to nor more than 93.5% in U-235

Authorized uses:

A. through L.

To be used for systems security checks; equipment standardization and calibration; process control; gauging and quality assurance testing; teaching; and nuclear reactor operations.

M. Primary reactor startup sources.

N. Reactor fuel

O. Excore neutron detectors

TABLE 12.2-21
REACTOR COOLANT SYSTEM DESIGN PARAMETERS - TO BE UPDATED BY WESTINGHOUSE

(Sheet 1 of 3)

Power Rating	
Total core thermal, MW	3565
Power density, W/cm	109.0
Effective Dimensions, in.	
Height	144.0
Diameter	132.7
Failed Fuel Fraction	
Equivalent fraction of average fuel rods containing cladding defects	0.01
Operation Times, Equivalent Full-Power Hours	
Initial cycle	9264
Equilibrium cycle	6480
Fission Product Escape Rate Coefficients During Full-Power Operation, sec^{-1}	
Noble gas isotopes	6.5×10^{-8}
Br, Rb, I, and Cs isotopes	1.3×10^{-8}
Te isotopes	1.0×10^{-9}
Mo isotopes	2.0×10^{-9}
Sr and Ba isotopes	1.0×10^{-11}
Y, Zr, Nb, La, Ce, and Pr isotopes	1.6×10^{-12}
Fission Product Escape Rate Coefficients During Spent Fuel Pit Storage, sec^{-1}	
Noble gas isotopes	6.5×10^{-13}
Br, Rb, I, and Cs isotopes	1.3×10^{-13}
Te isotopes	1.0×10^{-14}
Mo isotopes	2.0×10^{-14}
Sr and Ba isotopes	1.0×10^{-16}

TABLE 12.2-21
REACTOR COOLANT SYSTEM DESIGN PARAMETERS - TO BE UPDATED BY WESTINGHOUSE

(Sheet 2 of 3)

Y, Zr, Nb, La, Ce, and Pr isotopes	1.6×10^{-17}
Number of reactor coolant loops	4
System water volume, ft^3	12,000
System normal operating pressure, psia	2250
Average temperature in core, F	590
Transit Times, sec	
In core	0.75
Out of core	9.16
Total	9.91
Core Wetted Areas, Effective, in^2	
Zirconium	9.42×10^6
Stainless steel	6.09×10^6
Inconel	1.01×10^6
Out-of-core wetted area, Inconel, in^2	2.74×10^7
Coolant Velocity, ft/sec	
Core	16.1
Steam generator	19.6
Nominal Base Metal Release Rates, mg/dm^2-mo	
Zirconium	0.0
Stainless steel	0.5
Inconel	1.0
Coolant crud level, ppm	0.1
Permanent Crud Film, Nominal, mg/dm^2	
In core	50
Out of core	50
Transient Crud Layer, Nominal, mg/dm^2	
In core	50

TABLE 12.2-21
REACTOR COOLANT SYSTEM DESIGN PARAMETERS - TO BE UPDATED BY WESTINGHOUSE

(Sheet 3 of 3)

Out of core	50
Initial Boron Concentrations, ppm	
Initial cycle	805
Equilibrium cycle	1080
Pressurizer	
Liquid volume, nominal, ft^3	1080
Vapor volume, nominal, ft^3	720
Total wetted area, ft^2	1200
Spray line flow, gpm	1.0
Isotopic Stripping Fractions	
Noble gases	1.0
Other isotopes	0.0
10-Percent Step Load Transients	
Maximum surge rate in pressurizer, gpm	1720
Duration of transient, sec	40

TABLE 12.2-22
CHEMICAL AND VOLUME CONTROL SYSTEM DESIGN PARAMETERS - TO BE UPDATED BY WESTINGHOUSE

(Sheet 1 of 2)

Letdown rate, normal purification, gpm	75
Boron Removal Rates, ppm/day	
Initial cycle	2.06
Equilibrium cycle	3.96
Effective Deborating Demineralizer Cut-In	
Concentration, ppm	100
Mixed-Bed Demineralizer	
Resin volume, ft^3	30.0
Demineralizer Isotopic Decontamination Factors	
Noble gases	1.0
Y-90, Y-91, Cs-134, Cs-136, Cs-137, Mo-99	1.0
Other isotopes	10.0
Cation-Bed Demineralizer	
Effective flow, gpm	7.5
Resin volume, ft^3	30.0
Demineralizer Isotopic Decontamination Factors	
Noble gases	1.0
Y-90, Y-91, Cs-134, Cs-137	10.0
Other isotopes	1.0
Volume Control Tank	
Liquid volume, nominal, ft^3	160
Vapor volume, nominal, ft^3	240

TABLE 12.2-22
CHEMICAL AND VOLUME CONTROL SYSTEM DESIGN PARAMETERS - TO BE UPDATED BY WESTINGHOUSE

(Sheet 2 of 2)

Isotopic Stripping Fractions	
Kr-85	2.3×10^{-5}
Kr-85M	2.7×10^{-1}
Kr-87	6.0×10^{-1}
Kr-88	4.3×10^{-1}
Xe-131M	1.0×10^{-2}
Xe-133	1.6×10^{-2}
Xe-133M	3.7×10^{-2}
Xe-135	1.8×10^{-1}
Xe-135M	8.0×10^{-1}
Xe-138	1.0

TABLE 12.2-23
STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM DESIGN PARAMETERS - TO BE UPDATED BY WESTINGHOUSE

Primary to secondary side leakage rate, gpd	20
Partition Factors in the Steam Generator	
Iodine	0.01
Others	0.0
Partition Factors in the Condenser and Feedwater Heaters	
Iodine	0.0005
Others	0.0
Decontamination factors for the CPS	
Iodine	10
Cs, Rb	2
Tritium	1
Others	10
Decontamination Factors for the SGBPS Demineralizers	
Iodine	1000
Cs, Rb	2000
Tritium	1
Others	100,000
Resin charge and filter replacement per year	1

TABLE 12.2-24
DESIGN PARAMETERS FOR POST-ACCIDENT DOSE EVALUATIONS -
SHOULD BE UPDATED BY WESTINGHOUSE

Power Level	3,565 MWt
Three region equillbrium cycle core at end of life. The three regions have operated at a specific power of 40.00 MW/MTU for 300, 600, and 900 EFPD, respectively.	
Design Basis Accident	
Reactor coolant volume	11500 ft^3
Refueling water volume	50240 ft^3
Fraction of total core fission product inventory released	
Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01
Gap activity-release	
Fraction of total core fission product inventory released	
I-131	0.08
Kr-85	0.10
Other Iodines and Noble gases	0.05
Fuel handling accident	
Number of fuel rods in discharge region experiencing gap activity release	264
Release fractions are the same as for the total gap acitivity release accident	
Radial peaking factor	1.65
Decay interval between reactor shutdown and commencement of refueling operations (hours)	75

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 10)

1. NUREG-0017, Revision 1 release rates were used for the determination of airborne concentrations, unless otherwise noted.

2. Partition and plate-out factors are assumed to be considered in the source concentrations, which are based upon NUREG-0017, Revision 1.

3. Leakage from the CVCS is assumed to be the primary contributor to the radioactive airborne concentrations in the Safeguards and Auxiliary Buildings. Leakage from the RHRS is also included for shutdown conditions. Both systems are assumed to contain reactor coolant.

4. No purging of the Containment Buildings is assumed. A continuous containment vent for pressure relief equal to 10% of the maximum vent flow rate is assumed to relieve pressure buildup in the building due to thermal transients and air leakage (e.g., Instrument Air System) within the building.

5. Of the total yearly NUREG-0017 tritium activity release from each reactor unit, 1400 Ci is assumed to be released via gaseous pathways, and therefore contributes to the airborne concentrations in each unit. Tritium concentrations in the Containment Buildings during normal power operations are based upon 100% relative humidity and 120°F. The humidity in the containment atmosphere is assumed to contain reactor coolant tritium activity. For all other cases, tritium is conservatively treated as a noble gas, with a partition factor of 1.0 (no removal).

6. The leak rates provided for the auxiliary building in NUREG-0017 apply to both the Auxiliary and Safeguards Buildings, and are assumed to be equally distributed between the two buildings for each plant unit.

7. Building Volumes:

Containment Building (each unit)	2.985×10^6 ft^3
Auxiliary Building	2.0×10^6 ft^3

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 10)

Safeguards Building (each unit)	1.07×10^6 ft^3
Fuel Handling Building	1.3×10^6 ft^3
Turbine Building (each unit)	
Mezzanine	1.77×10^6 ft^3
Basement	1.72×10^6 ft^3
8. Ventilation Flow Rates:	
Containment Building (each unit)	
Purge	26,000 scfm (shutdown)
Pressure Relief	78.4 scfm (normal power operations)
Auxiliary Building	107,600 scfm
Safeguards Building (each unit)	20,200 scfm
Fuel Building	33,500 scfm
Turbine Building (each unit):	
Mezzanine	276,000 scfm (assumed)

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 10)

Basement 276,000 scfm

9. Release Rates for Particulates and Other Isotopes, Per Building, Per Unit, (Ci/year).

Isotope	Auxiliary[b] Building	Safeguards Building	Fuel Handling Building	Containment Building	Waste Gas System
Cs-134	3.03E-04	2.70E-04	1.70E-03	2.50E-03	3.30E-05
Cs-136	2.93E-05	2.40E-05	0.00E+00	3.20E-03	5.30E-06
Cs-137	4.37E-04	3.60E-04	2.70E-03	5.50E-03	7.70E-05
Cr-51	1.74E-04	1.60E-04	1.80E-04	9.20E-03	1.40E-05
Mn-54	4.11E-05	3.90E-05	3.00E-04	5.30E-03	2.10E-06
Fe-59	2.68E-05	2.50E-05	0.00E+00	2.70E-03	1.80E-06
Co-57	0.00E+00	0.00E+00	0.00E+00	8.20E-04	0.00E+00
Co-58	9.59E-04	9.50E-04	2.10E-02	2.50E-02	8.70E-06
Co-60	2.69E-04	2.55E-04	8.20E-03	2.60E-03	1.40E-05
Sr-89	4.19E-04	3.75E-04	2.10E-03	1.30E-02	4.40E-05
Sr-90	1.62E-04	1.45E-04	8.00E-04	5.20E-03	1.70E-05

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 4 of 10)

Sb-125	1.95E-06	1.95E-06	5.70E-05	0.00E+00	0.00E+00
Zr-95	5.05E-04	5.00E-04	3.60E-06	0.00E+00	4.80E-06
Isotope	Auxiliary[b] Building	Safeguards Building	Fuel Handling Building	Containment Building	Waste Gas System
Nb-95	1.87E-05	1.50E-05	2.40E-03	1.80E-03	3.70E-06
Ru-103	1.47E-05	1.15E-05	3.80E-05	1.60E-03	3.20E-06
Ru-106	5.70E-06	3.00E-06	6.90E-05	0.00E+00	2.70E-06
Ba-140	2.23E-04	2.00E-04	0.00E+00	0.00E+00	2.30E-05
Ce-141	1.52E-05	1.30E-05	4.40E-07	1.30E-03	2.20E-06
Other Isotopes, Per Unit					
C-14	3.45E+00	2.25E+00	0.00E+00	1.60E+00	1.20E+00
Ar-41			3.40E+01		

10. Primary and Secondary Coolant Concentrations (based upon NUREG-0017, Revision 1).

		Steam Generator Activity (uCi/g)	
Isotope	RCS Conc. (uCi/g)	Water	Steam

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 5 of 10)

Group I			
Kr-85m	1.39E-01		2.94E-08
Kr-85	1.37E-02		2.84E-09
		Steam Generator Activity (uCi/g)	
Isotope	RCS Conc. (uCi/g)	Water	Steam
Kr-87	1.53E-01		3.06E-08
Kr-88	2.61E-01		5.50E-08
Xe-131m	9.54E-02		1.96E-08
Xe-133m	2.70E-02		5.78E-09
Xe-133	5.90E-01		1.23E-07
Xe-135m	1.43E-01		2.98E-08
Xe-135	6.64E-01		1.41E-07
Xe-137	3.82E-02		7.98E-09
Xe-138	1.33E-01		2.76E-08
Group II			

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 6 of 10)

I-131	5.08E-02	2.19E-06	2.19E-08
I-132	2.37E-01	3.98E-06	3.98E-08
I-133	1.58E-01	5.92E-06	5.92E-08

Isotope	RCS Conc. (uCi/g)	Steam Generator Activity (uCi/g) Water	Steam
I-134	3.84E-01	3.13E-06	3.13E-08
I-135	2.94E-01	8.30E-06	8.30E-08
Group III			
Cs-134	4.28E-02	2.15E-06	1.11E-08
Cs-136	4.23E-03	2.10E-07	1.05E-09
Cs-137	5.70E-02	2.87E-06	1.44E-08
Group IV			
H-3	1.00E+00	1.00E-03	1.00E-03
Group V			
Cr-51	3.45E-03	1.56E-07	7.56E-10

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 7 of 10)

Mn-54	1.78E-03	7.79E-08	3.96E-10
Fe-59	3.34E-04	1.44E-08	7.32E-11
Co-58	5.12E-03	2.28E-07	1.13E-09

		Steam Generator Activity (uCi/g)	
Isotope	RCS Conc. (uCi/g)	Water	Steam
Co-60	5.90E-04	2.64E-08	1.32E-10
Sr-89	1.54E-04	6.77E-09	3.44E-11
Sr-90	1.32E-05	5.82E-10	2.97E-12
Zr-95	4.60E-02	2.03E-06	1.00E-08
Nb-95	2.08E-02	8.79E-07	4.56E-09
Ru-103	6.11E-01	2.72E-05	1.41E-07
Ru-106	2.45E+01	1.09E-03	5.28E-06
Ba-140	4.09E-01	1.76E-05	8.82E-08
Ce-141	1.05E-02	4.59E-07	2.33E-09

11. Containment Building Release Rates, (Each Reactor Unit)

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 8 of 10)

Noble Gases:	3% of RCS/day	(normal power operations)
Iodines:	8.0×10^{-4}%/day	(normal power operations)
	0.32 Ci/yr per uCi/g (RCS)	(shutdown / refueling)
Tritium:	18% of NUREG-0017 release, based upon 100% relative humidity and 120°F during normal power operations, and the remainder of the total yearly containment release during shutdown / refueling conditions.	
Particulates and Other Isotopes:	As noted previously.	

12. Auxiliary / Safeguards Buildings Release Rates (combined) for each reactor unit

Noble Gases:	160 lb/day	(normal power operations)
Iodines:	0.68 Ci/yr per uCi/g (RCS)	(normal power operations)
Iodines:	2.50 Ci/yr per uCi/g (RCS)	(shutdown / refueling)
Tritium:	32% of NUREG-0017 release.	
Particulates and Other Isotopes:	As noted previously.	

13. Worst Potential Auxiliary / Safeguards Room

Release Rates: 10% of the combined Auxiliary / Safeguards Buildings release rates, based upon relative leakage potential in Auxiliary and Safeguards Building rooms (normal power operations).

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 9 of 10)

7% of the combined Auxiliary / Safeguards Buildings release rates, based upon relative leakage potential in Auxiliary and Safeguards Building rooms (shutdown and refueling operations).

Volume: 1653 ft^3

Ventilation Flow: 200 scfm

14. Turbine Generator Building Release Rates from each reactor unit

Noble Gases: 1700 lb/hr of secondary steam

Iodines: 3800 Ci/yr per uCi/g of secondary steam (normal power operations)

Iodines: 420 Ci/yr per uCi/g of secondary steam (shutdown and refueling operations)

Tritium: 1700 lb/hr of secondary steam

Particulates and Other Isotopes: 1700 lb/hr of secondary steam

15. Fuel Handling Building Release Rates from each reactor unit

Noble Gases: Produced predominately from decay of iodines. Due to the high ventilation rates in the vicinity of the spent fuel pools, noble gas concentrations due to iodine decay are assumed to be negligible.

Iodines: 0.038 Ci/yr per uCi/g (RCS) (normal power operations)

Iodines: 0.093 Ci/yr per uCi/g (RCS) (shutdown / refueling)

TABLE 12.2-25
ASSUMPTIONS AND PARAMETERS USED IN DETERMINING AIRBORNE ACTIVITY CONCENTRATIONS - ORIGINAL LICENSING BASIS[a]

(Sheet 10 of 10)

Tritium:	32% of NUREG-0017 release.
Particulates and Other Isotopes:	As noted previously.

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

b) The CPNPP Auxiliary Building release rates are determined by adding 50% of the NUREG-0017 auxiliary building release rates to the Waste Gas System release rates from each reactor unit. This adjustment is made since the Waste Gas System at CPNPP is contained with the Auxiliary Building.

TABLE 12.2-26
RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 1 of 10)

(Normal Operations)

Isotope	Auxiliary Building	Each Safeguards Building	Each Containment Building	Fuel Building	Each Turbine Mezzanine	Each Turbine Basement	Worst Room
Kr-85m	2.19E-09	5.38E-09	3.03E-06		4.77E-14	1.19E-14	1.21E-07
Kr-85	2.27E-10	6.05E-10	2.96E-05		4.67E-15	1.17E-15	1.22E-8
Kr-87	2.16E-09	4.55E-09	9.54E-07		4.75E-14	1.19E-14	1.27E-07
Kr-88	4.02E-09	9.45E-09	3.58E-06		8.82E-14	2.21E-14	2.25E-07
Xe-131m	1.58E-09	4.20E-09	8.19E-05		3.23E-14	8.07E-15	8.50E-08
Xe-133m	4.45E-10	1.18E-09	6.35E-06		9.51E-15	2.38E-15	2.40E-08
Xe-133	9.76E-09	2.59E-08	2.87E-04		2.02E-13	5.04E-14	5.25E-07
Xe-135m	1.29E-09	1.86E-09	1.80E-07		3.80E-14	9.56E-15	9.29E-08
Xe-135	1.07E-08	2.75E-08	2.94E-05		2.30E-13	5.74E-14	5.85E-07
Xe-137	1.45E-10	1.59E-10	1.21E-08		6.09E-15	1.54E-15	1.37E-08
Xe-138	1.15E-09	1.63E-09	1.55E-07		3.46E-14	8.72E-15	8.42E-08
Ar-41	N/A	N/A	1.47E-07		N/A	N/A	N/A
I-131	2.62E-11	6.97E-11	8.97E-09	9.40E-12	2.47E-14	6.17E-15	1.41E-09
I-132	1.12E-10	2.58E-10	7.10E-10	3.68E-11	4.34E-14	1.09E-14	6.33E-09

TABLE 12.2-26

RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 2 of 10)

(Normal Operations)

Isotope	Auxiliary Building	Each Safeguards Building	Each Containment Building	Fuel Building	Each Turbine Mezzanine	Each Turbine Basement	Worst Room
I-133	8.09E-11	2.11E-10	4.13E-09	2.87E-11	6.64E-14	1.66E-14	4.38E-09
I-134	1.60E-10	3.12E-10	4.43E-10	4.72E-11	3.25E-14	8.14E-15	9.64E-09
I-135	1.47E-10	3.70E-10	2.51E-09	5.10E-11	9.23E-14	2.31E-14	8.05E-09
Cs-134	4.60E-13	1.09E-12	2.54E-09	8.30E-12	1.82E-14	4.55E-15	2.21E-11
Cs-136	4.45E-14	9.69E-14	1.38E-09		1.73E-15	4.32E-16	1.96E-12
Cs-137	6.64E-13	1.46E-12	5.73E-09	1.32E-11	2.37E-14	5.91E-15	2.94E-11
Cr-51	2.64E-13	6.47E-13	5.77E-09	8.78E-13	1.24E-15	3.11E-16	1.31E-11
Mn-54	6.24E-14	1.58E-13	5.22E-09	1.46E-12	6.51E-16	1.63E-16	3.19E-12
Fe-59	4.07E-14	1.01E-13	2.00E-09		1.20E-16	3.01E-17	2.04E-12
Co-57			8.01E-10				
Co-58	1.46E-12	3.84E-12	2.07E-08	1.02E-10	1.86E-15	4.64E-16	7.77E-11
Co-60	4.09E-13	1.03E-12	2.69E-09	4.00E-11	2.17E-16	5.43E-17	2.08E-11
Sr-89	6.37E-13	1.52E-12	9.94E-09	1.02E-11	5.67E-17	1.42E-17	3.07E-11
Sr-90	2.46E-13	5.87E-13	5.41E-09	3.90E-12	4.89E-18	1.22E-18	1.19E-11

TABLE 12.2-26
RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 3 of 10)

(Normal Operations)

Isotope	Auxiliary Building	Each Safeguards Building	Each Containment Building	Fuel Building	Each Turbine Mezzanine	Each Turbine Basement	Worst Room
Sb-125	2.96E-15	7.89E-15		2.78E-13	1.59E-13		
Zr-95	7.67E-13	2.02E-12		1.76E-14	1.65E-14	4.13E-15	4.09E-11
Nb-95	2.84E-14	6.07E-14	1.23E-09	1.17E-11	7.50E-15	1.87E-15	1.23E-12
Ru-103	2.23E-14	4.65E-14	1.14E-09	1.85E-13	2.31E-13	5.79E-14	9.40E-13
Ru-106	8.66E-15	1.21E-14		3.37E-13	8.70E-12	2.17E-12	2.45E-13
Ba-140	3.39E-13	8.08E-13			1.45E-13	3.63E-14	1.63E-11
Ce-141	2.31E-14	5.26E-14	8.67E-10	2.15E-15	3.84E-15	9.60E-16	1.06E-12
C-14	5.24E-09	9.11E-09	1.67E-06		1.84E-07		
H-3	3.37E-07	8.97E-07	7.85E-05	2.36E-06	1.65E-09	4.11E-10	1.81E-05
Total[b] MPC	9.01E-02	2.32E-01	7.40E+01	4.88E-01	1.80E-03	4.49E-04	4.90E+00

TABLE 12.2-26
RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 4 of 10)

(Shutdown and Refueling Operations)

Isotope	Each Auxiliary Building	Each Safeguards Building	Each Containment Building	Each Fuel Building	Turbine Mezzanine	Turbine Basement	Worst Room
I-131	4.45E-10	1.18E-09	2.34E-10	1.06E-10	1.26E-14	3.15E-15	1.67E-08
I-132	1.90E-09	4.37E-09	6.98E-10	4.15E-10	2.22E-14	5.54E-15	7.50E-08
I-133	1.37E-09	3.59E-09	6.91E-10	3.24E-10	3.39E-14	8.47E-15	5.18E-08
I-134	2.71E-09	5.29E-09	7.11E-10	5.32E-10	1.66E-14	4.15E-15	1.14E-07
I-135	2.49E-09	6.28E-09	1.14E-09	5.75E-10	4.71E-14	1.18E-14	9.53E-08
H-3	4.45E-10		2.53E-06	4.72E-06			
Total							
MPC[b]	1.35E-01	3.46E-01	5.71E-01	9.75E-01	3.14E-06	7.86E-07	5.14E+00

TABLE 12.2-26

RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 5 of 10)

(1 Unit in Normal Operations, 1 Unit Shutdown or Refueling)

Isotope	Auxiliary Building	Normal Power Unit Safeguards Building	Shutdown/ Refueling Unit Safeguards Building	Normal Power Unit Reactor Building	Shutdown/ Refueling Unit Reactor Building	Fuel Building
Kr-85m	1.09E-09	5.38E-09		3.03E-06		
Kr-85	1.14E-10	6.05E-10		2.96E-05		
Kr-87	1.08E-09	4.55E-09		9.54E-07		
Kr-88	2.01E-09	9.45E-09		3.58E-06		
Xe-131m	7.89E-10	4.20E-09		8.19E-05		
Xe-133m	2.22E-10	1.18E-09		6.35E-06		
Xe-133	4.88E-09	2.59E-08		2.87E-04		
Xe-135m	6.44E-10	1.86E-09		1.80E-07		
Xe-135	5.37E-09	2.75E-08		2.94E-05		
Xe-137	7.26E-11	1.59E-10		1.21E-08		
Xe-138	5.76E-10	1.63E-09		1.55E-07		
Ar-41	N/A	N/A		1.47E-07		
I-131	2.36E-10	6.97E-11	1.18E-09	8.97E-09	2.34E-10	5.77E-11
I-132	1.01E-09	2.58E-10	4.37E-09	7.10E-10	6.98E-10	2.26E-10

TABLE 12.2-26
RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 6 of 10)

(1 Unit in Normal Operations, 1 Unit Shutdown or Refueling)

Isotope	Auxiliary Building	Normal Power Unit Safeguards Building	Shutdown/ Refueling Unit Safeguards Building	Normal Power Unit Reactor Building	Shutdown/ Refueling Unit Reactor Building	Fuel Building
I-133	7.27E-10	2.11E-10	3.59E-09	4.13E-09	6.91E-10	1.76E-10
I-134	1.43E-09	3.12E-10	5.29E-09	4.43E-10	7.11E-10	2.90E-10
I-135	1.32E-09	3.70E-10	6.28E-09	2.51E-09	1.14E-09	3.13E-10
Cs-134	2.30E-13	1.09E-12		2.54E-09		4.15E-12
Cs-136	2.22E-14	9.69E-14		1.38E-09		
Cs-137	3.32E-13	1.46E-12		5.73E-09		6.59E-12
H-3	1.68E-07	8.97E-07		7.85E-05	2.53E-06	3.54E-06
Cr-51	1.32E-13	6.47E-13		5.77E-09		4.39E-13
Mn-54	3.12E-14	1.58E-13		5.22E-09		7.32E-13
Fe-59	2.04E-14	1.01E-13		2.00E-09		
Co-57				8.01E-10		
Co-58	7.28E-13	3.84E-12		2.07E-08		5.12E-11
Co-60	2.04E-13	1.03E-12		2.69E-09		2.00E-11
Sr-89	3.18E-13	1.52E-12		9.94E-09		5.12E-12

TABLE 12.2-26

RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 7 of 10)

(1 Unit in Normal Operations, 1 Unit Shutdown or Refueling)

Isotope	Auxiliary Building	Normal Power Unit Safeguards Building	Shutdown/ Refueling Unit Safeguards Building	Normal Power Unit Reactor Building	Shutdown/ Refueling Unit Reactor Building	Fuel Building
Sr-90	1.23E-13	5.87E-13		5.41E-09		1.95E-12
Sb-125	1.48E-15	7.89E-15				1.39E-13
Zr-95	3.83E-13	2.02E-12				8.78E-15
Nb-95	1.42E-14	6.07E-14		1.23E-09		5.85E-12
Ru-103	1.12E-14	4.65E-14		1.14E-09		9.27E-14
Ru-106	4.33E-15	1.21E-14				1.68E-13
Ba-140	1.69E-13	8.08E-13				
Ce-141	1.15E-14	5.26E-14		8.67E-10		1.07E-15
C-14	2.62E-09	9.11E-09		1.67E-06		0.00E+00
Total[b] MPC:	1.13E-01	2.32E-01	3.46E-01	7.40+01	5.71E-01	7.32E-01

TABLE 12.2-26
RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 8 of 10)

(1 Unit in Normal Operations, 1 Unit Shutdown or Refueling)

Isotope	Normal Power Unit Turbine Mezzanine	Shutdown/ Refueling Unit Turbine Mezzanine	Normal Power Unit Turbine Basement	Shutdown/ Refueling Unit Turbine Basement	Normal Power Unit Worst Room	Shutdown/ Refueling Unit Worst Room
Kr-85m	4.77E-14		1.19E-14		1.21E-07	
Kr-85	4.67E-15		1.17E-15		1.22E-08	
Kr-87	4.75E-14		1.19E-14		1.27E-07	
Kr-88	8.82E-14		2.21E-14		2.25E-07	
Xe-131m	3.23E-14		8.07E-15		8.50E-08	
Xe-133m	9.51E-15		2.38E-15		2.40E-08	
Xe-133	2.02E-13		5.04E-14		5.25E-07	
Xe-135m	3.80E-14		9.56E-15		9.29E-08	
Xe-135	2.30E-13		5.74E-14		5.85E-07	
Xe-137	6.09E-15		1.54E-15		1.37E-08	
Xe-138	3.46E-14		8.72E-15		8.42E-08	
Ar-41	N/A		N/A		N/A	
I-131	2.47E-14	1.26E-14	6.17E-15	3.15E-15	1.41E-09	1.67E-08
I-132	4.34E-14	2.22E-14	1.09E-14	5.54E-15	6.33E-09	7.50E-08

TABLE 12.2-26
RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 9 of 10)

(1 Unit in Normal Operations, 1 Unit Shutdown or Refueling)

Isotope	Normal Power Unit Turbine Mezzanine	Shutdown/ Refueling Unit Turbine Mezzanine	Normal Power Unit Turbine Basement	Shutdown/ Refueling Unit Turbine Basement	Normal Power Unit Worst Room	Shutdown/ Refueling Unit Worst Room
I-133	6.64E-14	3.39E-14	1.66E-14	8.47E-15	4.38E-09	5.18E-08
I-134	3.25E-14	1.66E-14	8.14E-15	4.15E-15	9.64E-09	1.14E-07
I-135	9.23E-14	4.71E-14	2.31E-14	1.18E-14	8.05E-09	9.53E-08
Cs-134	1.82E-14		4.55E-15		2.21E-11	
Cs-136	1.73E-15		4.32E-16		1.96E-12	
Cs-137	2.37E-14		5.91E-15		2.94E-11	
H-3	1.65E-09		4.11E-10		1.81E-05	
Cr-51	1.24E-15		3.11E-16		1.31E-11	
Mn-54	6.51E-16		1.63E-16		3.19E-12	
Fe-59	1.20E-16		3.01E-17		2.04E-12	
Co-57					7.77E-11	
Co-58	1.86E-15		4.64E-16		2.08E-11	
Co-60	2.17E-16		5.43E-17		3.07E-11	
Sr-89	5.67E-17		1.42E-17		1.19E-11	

TABLE 12.2-26
RADIOACTIVE AIRBORNE CONCENTRATIONS (uCi/cc) - ORIGINAL LICENSING BASIS[a]

(Sheet 10 of 10)

(1 Unit in Normal Operations, 1 Unit Shutdown or Refueling)

Isotope	Normal Power Unit Turbine Mezzanine	Shutdown/ Refueling Unit Turbine Mezzanine	Normal Power Unit Turbine Basement	Shutdown/ Refueling Unit Turbine Basement	Normal Power Unit Worst Room	Shutdown/ Refueling Unit Worst Room
Sr-90	4.89E-18		1.22E-18		1.59E-13	
Sb-125	4.09E-11					
Zr-95	1.65E-14		4.13E-15		1.23E-12	
Nb-95	7.50E-15		1.87E-15		9.40E-13	
Ru-103	2.31E-13		5.79E-14		2.45E-13	
Ru-106	8.70E-12		2.17E-12		1.63E-11	
Ba-140	1.45E-13		3.63E-14		1.06E-12	
Ce-141	3.84E-15		9.60E-16		1.84E-07	
C-14						
Total[b] MPC	1.80E-03	3.14E-06	4.49E-04	7.86E-07	4.90E+00	5.14E+00

a) Historical. Not subject to future updating. Has been retained to preserve original design basis.

b) Design assessment made with the provision of 10 CFR 20.1-20.601.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

The design guidance given in NRC Regulatory Guide 8.8, Revision 3, Section C.2 was applied in the facility and equipment design, as practicable, consistent with benefits received and costs incurred. The following features are provided in the design of equipment and development of plant arrangements.

12.3.1.1 Equipment Design Features

The following design features are provided for the nuclear system equipment that contains sources of radiation to ensure that ORE's are ALARA.

12.3.1.1.1 Evaporators

1. Evaporator packages are designed with consideration given to occupational exposure. A 24-in. thick concrete shield separates the evaporator concentrator tank, gas stripper, and vent condenser from the pumps, valves, and instrumentation. The 24-in. thick concrete provides significant dose reductions. The systems are skid-mounted and arranged so that the evaporator components are easily accessible.

2. Instruments installed in lines which service the evaporators are located outside the evaporator compartments.

3. Shielding is provided for local sample vessels associated with evaporator streams.

12.3.1.1.2 Pumps, Filters, and Demineralizers

1. Canned pumps are used, where appropriate in the WPS to eliminate the potential for leaks. In addition, pumps which carry concentrated boric acid are supplied with high-quality mechanical seals to minimize pump maintenance.

2. Where it is possible, NSSS pumps are provided with flanges to facilitate removal and replacement.

3. Canned pump stators are air-filled, as opposed to silicon oil-filled, to eliminate the potential for process stream contamination and maintenance.

4. In cases where pumps are located in shielded compartments and local start stop or jog switches are required, these switches are located outside the pump compartment.

5. Filters are designed for removal from the top, using a minimum number of tools. Filter bolt lead-in for tool entry is provided to facilitate remote loosening. A detailed description of expended filter cartridge processing is provided in Section 11.4.2.

6. It is possible to remove filter cartridge assemblies into self-shielded transfer containers without the requirements of visual observation; see Section 11.4.2.

7. All demineralizers in the NSSS are considered potentially radioactive. Therefore, each demineralizer is located inside an individually shielded compartment except the steam generator cation blowdown demineralizers and mixed bed blowdown demineralizers, which are grouped and shielded by function and unit.

8. To minimize personnel exposure, special care was taken to ensure that only the required process lines enter into or pass through the demineralizer cubicles.

9. Design changes have been incorporated in the reactor coolant pump seals to improve reliability/maintainability. The No. 2 and No. 3 seal consists of a factory tested and assembled cartridge seal assembly. This unit reduces the time required for seal changeout and maintenance. It is projected the RCP seal life is approximately 4.5 years as opposed to previous 3 year time frame. Thus, radiation exposures will be reduced due to the quicker change time and the reduced frequency of this maintenance.

   An improved reactor coolant pump maintenance system has been developed which allows shaft seal access without removing the pump motor. This reduces manpower, radiation exposure, and pump maintenance downtime.

10. The use of a refueling cavity recirculation cleanup system, equipped with filters, will minimize radioactive particulates in the cavity shield water. This filtration reduces the radiation level at the surface of the water and reduces contamination of the cavity walls upon cavity draining at completion of refueling. Thus, radiation exposure to personnel during cavity cleaning is reduced.

12.3.1.1.3 Tanks and Heat Exchangers

1. To facilitate draining, horizontal and flat-bottom tanks are normally sloped toward the drain, thus allowing radioactive sediment to be more easily flushed from the tank, resulting in lower exposures to plant personnel.

2. To prevent radioactive gases and vapors from escaping to the atmosphere via tank overflows, loop seals are provided, where appropriate.

3. NSSS heat exchangers are designed without the need for special maintenance fixtures, such as blind flanges, which add to maintenance time. For example, by threading every fourth hole in the heat exchanger tube sheet, the shell-to-tube sheet joint need not be broken for maintenance and inspection. The shell and tube assembly can be lifted intact above the channel head to expose the tube ends for inspection or testing, or both, for leaks.

12.3.1.1.4 Valves

1. Special attention is given to valve locations so that operation and maintenance is conveniently performed from standard service equipment such as ladders, platforms, and floors.

2. All modulating and three-way control valves which normally carry radioactive fluid have leakoffs.

3. Bolted body-to-bonnet forgings are used. This design permits use of ultrasonic testing in place of time-consuming radiography during inservice inspection. In addition, this design facilitates assembly/disassembly, and eliminates the need for gaskets as required by the pressure seal valve design. On two in. Pipes and smaller, most valves are designed for zero stem leakage.

12.3.1.1.5 Piping

1. Where possible, piping is arranged so that ample space is available for installing and maintaining electrical heat-tracing equipment, thus reducing jobtime requirements and exposure.

2. Pipe runs carrying radioactive fluid are arranged so that potential crud traps are minimized.

3. Where possible, for lines which are used to transport spent resin, six-diameter bends and continually sloping lines are used.

4. The number of vents and drains on piping are minimized to reduce possible leak paths and likely crud traps.

12.3.1.1.6 Reactor Coolant Piping Insulation

Insulation in the areas of the reactor vessel nozzle welds is fabricated sections with a thin stainless sheet covering, featuring “suitcase” type clasps, to hold the sections in place. By merely unsnapping four, a matter of minutes time, a 29-in.-diameter pipe weld can be exposed for examination. Replacing the insulation after the job is as fast, thereby significantly reducing exposure time.

12.3.1.1.7 Process Systems

Concepts such as minimizing radioactive wastes and thus reducing radiation exposure were realized through process design techniques. Water reuse designs will significantly reduce the quantity of water processed in the liquid radwaste system. By virtue of processing less radioactive liquids, a reduced volume of evaporator bottoms is generated, thereby reducing potential exposure due to handling in the solid waste management system. The design for reuse of resin sluice water is illustrated on Figure 12.3-1. Water pumped from the spent resin storage tank into demineralizers transports the refueling water/resin slurry back to the tank. The sluice water is then reused after passing through a resin retention screen within the tank. Another means of minimizing radioactive effluent handling was to incorporate a boron thermal regeneration system (BTRS) [1]. Effluents generated during load follow operations with BTRS range from 2 to 10 system volumes per year compared to 50 to 60 system volumes per year for plants without BTRS.

Arrangement of unit operations in the system design precludes excess exposure which might result from subsequent maintenance operations. Demineralization and filtration of process streams prior to entry into large storage tanks is one method of equipment arrangement employed. The objective is to retard activity accumulation within the tanks thereby minimizing the source of radiation. In the BRS, the recycle evaporator feed demineralizers are located upstream from the recycle holdup tanks. The prime reason for this location is to minimize the

activity concentration of the effluent entering the tanks and thereby keep the tanks as clean as possible. Table 12.3-1 gives demineralizer decontamination factors (DF) for various isotopes measured in two operating plants [2]. These demineralizers are used for reactor coolant purification. Decontamination factors of similar magnitude are expected for the recycle evaporator feed demineralizers which contain resin in the hydrogen hydroxyl form.

12.3.1.1.8 Refueling

The design of refueling components incorporates such improved features as:

1. An integral missile shield (Unit 1 only) and a roll-away missile shield (Unit 2)

2. Quick disconnect cables for CRDM's

3. Elimination of part-length CRDM's

4. A permanent reactor cavity seal ring

5. A quick reactor vessel head removal system

6. An upgraded manipulator crane control system

7. In-Containment fuel storage

8. A Containment fuel handling bridge crane

9. A reactor vessel head shield.

These refueling features can considerably reduce the time required to perform refueling operations and, consequently, the overall downtime for reactor refueling. These features are designed with consideration given to reducing ORE; however, they cannot be economically justified solely on the basis of reduced ORE.

The pressure vessel head closure system is designed to reduce the time required to tension and detention the reactor vessel studs and to remove and insert the studs into the vessel flange. The system includes quick disconnect/connect stud tensioners which have split, quick acting stud gripper devices to eliminate the need for threading tensioners onto the studs. The system also includes pneumatic tools for removing/inserting studs into the vessel flange.

Connectors on electrical cables permit easy disconnect/connect of the cables when the reactor vessel head is removed, thereby reducing the time spent in making and verifying electrical connections following refueling. A cable tray also permits convenient storage of these cables. These developments significantly reduce personnel exposure during refueling operations.

An integral TV camera that has been designed into the upgraded refueling machine (manipulator crane) or a portable camera may be used for visual confirmation of positioning during fuel handling. This camera or a portable camera is to be used for core-mapping. The speed and accuracy of the refueling machine movement and positioning has also been improved. The features incorporated into the upgrade refueling machine serve to reduce both the refueling time and the radiation exposure associated with refueling operations.

Since the occupational radiation exposures for refueling vary from shutdown to shutdown at a given plant and vary among different plants, no single estimate can be established on the savings in occupational radiation exposure attributable to the improved refueling concept. However, an estimate may be obtained by using data from a typical refueling which usually results in radiation exposures ranging from 30 to 50 man-rem. It is estimated that the improved refueling concept at CPNPP could reduce these values to the range of 10 to 25 man-rem.

12.3.1.2 Facility Design Considerations

The following considerations are incorporated into the arrangement and design of the facility to ensure that occupational radiation exposures are ALARA:

12.3.1.2.1 Maintenance Considerations

The layout of personnel access areas, pipe routing, and equipment arrangement minimize occupational radiation exposure during maintenance activities. Radioactive equipment requiring service is normally located in individually shielded cubicles of a size designed to provide easy accessibility for maintenance. In addition, adequate equipment-handling facilities, such as cranes and overhead rails, are provided for maintenance work.

Accessibility for equipment removal was considered during the plant design so that component removal for servicing is possible. Access control and traffic patterns are considered in the plant layout to minimize the spread of contamination. The following design features are provided to minimize the spread of contamination:

1. Cubicles containing components such as pumps, heat exchangers, and valve stations are provided with vent and drain systems. The equipment vents and drains are piped to a collection system to prevent any contaminated fluid from flowing across the floor of the cubicle.

2. All-welded piping systems are used in radioactive systems to the maximum extent possible to reduce system leakage.

3. Valves designated for radioactive system service are normally provided either with leakoff connections piped directly to a collection system or with diaphragm or bellow seals. Valve designs without stem seals or leakoffs are used when ALARA considerations are accomplished by other means.

4. Tank cubicles are provided with raised entrances to prevent the spread of contamination in the event of a tank rupture.

For maintenance purposes where equipment access is required, various equipment such as pumps and heat exchangers can be remotely flushed before direct maintenance occurs by personnel.

The capability to decontaminate potentially contaminated areas is facilitated by the availability of decontamination water hose stations and by the use of a suitable concrete-finish surface design. Protective coating materials used in radiation areas were originally selected using ANSI N512-1974 and the coating manufacturer's recommendations for guidance to provide decontaminable finishes. See Section 6.1B.2 for a description of the selection and qualification of protective

coatings. Floor drains are provided in all areas of the plant that are subject to potential contamination.

Systems which may become contaminated are normally designed for local decontamination of components prior to servicing. The capability of performing local decontamination is provided by a system design that incorporates proper slopes for drainage, proper vents at high points, drains at low points, and valving for major equipment isolation.

12.3.1.2.2 Penetrations

Because precautions are taken to minimize the contribution of radiation streaming through penetrations, penetrations through shield walls or slabs for pipes, extension stems, ventilation ducts, and cable trays are located where possible so that they are not in a direct line with major radiation sources. In addition, bends and offsets are used when required. Also, filling materials (such as lead wool) are used where applicable to fill voids in penetrations.

Separate shielded areas in the Safeguard Buildings are provided for radioactive and nonradioactive Containment piping penetrations.

12.3.1.2.3 Remote Handling Equipment

In accordance with NRC Regulatory Guide 8.8, Revision 3, remote handling equipment and other special tools are provided, when necessary, to reduce external radiation exposure. Careful design in conjunction with proper use of equipment minimizes radiation exposures to plant personnel. Where necessary in areas of high radioactivity, provisions, such as stem extensions through shielding walls and slabs, are made to reduce personnel exposure. Specifically, remote handling equipment is used for the following:

1. Remote handling equipment is used for waste processing in the Fuel Building. A description of the solid waste management system is given in Section 11.4.

2. Remote handling equipment is used in the in-core instrumentation room where the flux mapping system is used to drive the neutron flux detectors into and out of the reactor core through the flux thimbles. This system consists mainly of drive mechanisms, path transfer devices, and a movable frame assembly.

3. Remote handling equipment is also used for transfer of spent filter cartridges to a solid waste disposal area in the Fuel Building. Filters considered to be potentially radioactive are grouped together and located inside individual shielded compartments. Provisions are made for safe handling of replaced filter cartridges and the spent filter cartridges are transported to the solid waste disposal area via a shielded filter element drop zone. A description of the handling operation is in Section 11.4.

12.3.1.2.4 Ventilation System

The ventilation system is designed to ensure control of airborne contaminants during all modes of operation including maintenance activities. Those aspects of the design that relate to controlling the concentration of airborne radioactivity in equipment cubicles, corridors, and operating areas which are normally occupied by operating personnel are discussed in Section 9.4.

The ventilation system is designed for easy access and service to keep doses ALARA during maintenance and filter changes. The design provisions that are included to achieve ALARA exposures in these situations are discussed in Subsection 12.3.3.

12.3.1.2.5 Tool Design for Maintenance and Inspection

Tools and procedures have been developed which are aimed at keeping ORE's ALARA during maintenance and inservice inspection tasks. The major emphasis is in areas with the potential to prevent large accumulations of radiation exposure, as described in the following paragraphs.

12.3.1.2.5.1 Inservice Inspection

1. Steam Generator Tubing

Periodic inservice inspection of the steam tubing is required to monitor the integrity of the tubing and to maintain surveillance of potential mechanical degradation or tube deterioration throughout the life of the steam generator. Examination via eddy current testing is the most practical technique. It was developed so that accurate data can be obtained. This technique requires that a long probe be inserted in a tube and then withdrawn at a constant rate of speed, during which time the signal is read and recorded.

Until recently, this technique required that a man stand at the open steam generator manway with a length of aluminum conduit and hold it in such a position that when an eddy current probe was pushed through the conduit, it would then enter the steam generator tube for the examination. This procedure would take as long as three to five minutes for examination of one tube, with the operator standing in a field of radiation which might range from 150 mrem/hr to 800 mrem/hr. To reduce exposures from this source, a fixture is used which remotely positions the probe in any desired tube and mechanically inserts and withdraws the probe. The control of the total examination is done at a remote location within the Containment, with the operators in a low field of radiation (typically 10 mrem/hr). This fixture does require the entry of individuals into the steam generator for short periods of time for installation and removal. However, exposure is significantly less than that using the hand probing method.

2. Reactor Vessel and Piping Inspection

Selected welds in the RCS of a NSSS must be examined periodically throughout the life of the plant to ensure the continued integrity of the welds. This examination is done by ultrasonic testing and dye penetrant techniques in close proximity to the component. Of special interest in the periodic examinations are the welds which attach the reactor coolant pipe nozzles to the reactor vessel. The examination is frequently complicated by the fact that most of the welds are covered by thick layers of thermal insulation which must be removed prior to the examination and replaced upon completion of the inspection.

A dual approach is taken to minimize radiation exposures during these inspections. The first provides an optimum design configuration of the component to readily accommodate the UT sensor. For example, the reactor vessel nozzle area is tapered along the reinforced areas to ensure a smooth transition. The branch pipe locations are selected to

ensure no interference from one branch to the next. All weld-to-pipe interfaces require a smooth, high-quality finish.

The second approach for reducing exposures is the use of remote inspection techniques. An inspection tool has been developed for examining the reactor vessel nozzle welds by positioning the ultrasonic crystals remotely and by providing readout of the signals in a low radiation area. This tool is inserted in the open reactor vessel after removal of the fuel and reactor vessel internals. The refueling cavity is flooded with water so that the operators are shielded by about 30 ft of water.

12.3.1.2.5.2 Maintenance and Repair Operations (Steam Generator Tube Plugging)

Occupational Radiation Exposure reduction is considered in the design of components for the steam generator tube plugging process, through maximizing the use of remote mechanical tube plugging methods.

12.3.1.2.6 Materials Selection and Manufacturing Process

Materials selection and manufacturing processes in the design and manufacture of components in direct contact with the reactor coolant can have a profound effect on radiation levels and subsequent potential radiation exposures. The two major contributors to radiation levels in the pressurized water reactor (PWR) are Co-58 and Co-60. Co-58 is formed from an (n,p) reaction on Ni-58, which is present in many alloys in contact with the RCS, particularly the Inconel steam generator tube material. Co-60 is formed from an (n, γ) reaction on Co-59 which is an impurity in stainless steel and Inconel. An additional potential source of Co-59 is Stellite (Registered trademark of Haynes Stellite Company), which is approximately 50 percent cobalt and is used on wearing surfaces.

The mechanism of accumulating deposits of radioactive materials in components involves the following steps [4]:

1. Release of base metal to the RCS

2. Deposition of base metal in active core region

3. Activation of base metal

4. Release of activated metal from core region

5. Deposition of activated metal in components

Any steps taken during the design and operation of a PWR to reduce any of the aforementioned actions will potentially reduce radiation levels in components.

Four areas related to materials selection and manufacturing processes are as follows:

1. Material selection to minimize maintenance

2. Bright annealing of steam generator tubes

3. Equipment specifications on cobalt content of materials

4. Chemistry control to minimize radiation levels and corrosion

1. Materials Selection to Minimize Maintenance

Reduced occupational exposures in operating PWRs can best be accomplished by the minimization of maintenance, repairs, and inspection.

a. Steam Generator Tubing

For a number of years a program has attempted to evaluate alternate steam generator materials. The program consists of various tests simulating steam generator conditions for a number of materials. For the Unit 1 replacement steam generators (RSGs), experience indicates that thermally-treated Inconel 690 is a superior material and it was selected for this unit. For the Unit 2 steam generators, it was concluded that Inconel 600 was one of the best steam generator materials for maintaining the integrity of the steam generator tubes in steam generators.

An additional consideration related to steam generator material selection is the quantity of base metal that can be activated and can result in subsequent radiation exposure. If the corrosion products released from two materials of different nickel content had the same composition as the base metal, then the material with the lower nickel content would result in the production of less Co-58 than the material with the higher nickel content. However, crud from operation reactor plants with Inconel 600 steam generators has been found to consist of nickel ferrite, Ni_xFe_{3-x} O4, with x less than or equal to 1.0. Thus, it is concluded that the released corrosion products from Inconel 600 have less nickel than base metal. If the crud composition for lower nickel content materials is the same, no benefit in Co-58 radioactivity buildup is found. In addition, there is no evidence of nickel in other compounds in core crud [4].

b. Hard Facing Materials

It has been suggested that consideration be given to Colmonoy and Hastelloys as replacements for Stellite as a means of potentially reducing the cobalt in crud. However, the choice of materials should be based on the best combination of superior wear characteristic or reduced inspection and maintenance requirements, or both, and potential contribution to inplant radiation exposure. With regard to Hastelloys, our experience shows that these materials do not have the required wear characteristics, even in cold worked condition, for our applications. Colmonoy has been investigated for possible application in components with the following conclusions:

1. Experience is with Colmonoy 5 (casting) and Colmonoy 6 (powder) weld deposited by oxyacetylene and plasma transferred arc with the latter being superior in the quality of the finished product.

2. Colmonoy thrust shoes undergo unacceptable corrosion attack in contact with Graphitar during wet layup. Thus, all thrust shoes continue to be made from Stellite.

3. Plasma hardsurfacing with Colmonoy requires wider base metal surface because of the increased fluidity as compared to Stellite.

4. Present Colmonoy hardsurfacing procedures require processing in the sensitization temperature range of austenitic stainless steels (900 to 1200°F).

5. Corrosion performance of the Colmonoy materials has not been demonstrated in borated water systems.

In addition, it should be noted that if these substitute materials have lesser wear characteristics compared to Stellite, the quantity of nickel introduced into the circulating coolant from these substitute materials could increase drastically. This would result in an increase in Co-58 levels in the plant.

Stellite was found to be a major contributor to cobalt in the Shippingport Atomic Station [5]. However, this reactor had check valves and stop valves (among others) in the active RCS, which contributed heavily to the cobalt contribution. Additionally, core crud deposits in the Shippingport reactor showed approximately 0.4 w/o cobalt.

(See the following discussion for significance of the cobalt content of core crud deposits.)

A review of the Stellite inventory in the NSSS indicates that there are four major areas of use: reactor internals, RCS pump journals, RCS valves, and CRDMs. The Stellite application in these areas is limited to load bearing surfaces in which excessive wear might otherwise result. A summary of the surface area wetted by the reactor coolant of these three applications is given in Table 12.3-2.

Figure 12.3-2 provides justification that Stellite is not a major contributor since the cobalt weight percent in crud is generally well below the 0.1-percent level. As a final point on this subject, tungsten is a component material in the Stellite alloy (Stellite has 45 percent tungsten). If significant contributions to the deposited radiation source term were originating from wear of Stellite material, a measurable radioactive tungsten level would be detected in the crud deposits. W-187 has only been detected in one or two reactors, but the level is approximately 40 times that which could be expected based on Co-59 measurements. Thus, the RCS W-187 is due to other sources in the system (i.e., Inconel impurity). Through implementation of a cobalt reduction program, newly developed low cobalt or cobalt-free iron based alloys which have been industry tested to demonstrate similar corrosion and wear characteristics as the Stellite family of hardfacing alloys are considered, primarily for use in RCS valves and other system valves which communicate with the reactor coolant system.

2. Bright Annealing of Steam Generator Tubes

Another technique which demonstrates efforts to reduce the source of occupational exposure is bright annealing of steam generator tubes. Bright annealing involves the

annealing of the Inconel tubes while both the inside and outside tube surfaces are exposed to a hydrogen atmosphere.

This process reduces the corrosion tendency of steam generator tubes and also minimizes the crud retention properties of the tubes. This technique has been used on all steam generators with Inconel tubes dating back to the Inconel 600 tubes used in the Zorita steam generators. The Connecticut-Yankee and San Onofre steam generators are the only ones using Inconel 600 which were not bright annealed. The tubes were grit blasted with $A1_2O_3$ to minimize crud retention properties.

3. Equipment Specifications on Cobalt Content of Materials

The current limits on cobalt content of materials in contact with reactor coolant are given in Table 12.3-3. The cobalt content of materials wetted by reactor coolant is restricted, although there have been infrequent occasions in the past when the restrictions were temporarily relaxed due to unavailability of material with the specified cobalt content. However, this relaxation has been for minor components of minimal surface area. Through implementation of a cobalt reduction program, specifications for valves which potentially contribute to the cobalt inventory reaching the reactor core have been updated to add consideration for the procurement of low cobalt or cobalt-free materials for replacement valves and parts. Such materials should have mechanical and physical characteristics suitable for valve functionality.

To evaluate the adequacy of these equipment specifications, input must be available regarding the actual cost of materials at various cobalt levels and the total cost benefit evaluation of materials at various cobalt levels:

a. Actual Cost of Materials at Various Cobalt Levels

The estimated additional costs for 304 stainless steel plates and forgings for various cobalt levels are given in Table 12.3-4. These values were prepared as part of the CPNPP Operating License application prior to Unit 1 operation and will not be updated. If the entire nuclear industry decided to order low cobalt <0.1-w/o stainless steel for all RCS components, the supply and demand would shift dramatically, thereby resulting in higher costs for reduced cobalt. The potential exists for using carbon steel clad internally with low cobalt stainless steel as an alternative to all low cobalt stainless steel. This potential has been investigated with the conclusion that the cost differential based on using carbon steel is offset by the cladding process costs, special welding costs, and so forth.

The equipment specification for Inconel 600 used in manufacturing steam generator tubes of 0.1-w/o cobalt results in field delivered tubes with cobalt contents ranging from 0.05-w/o to 0.1-w/o cobalt. The average cobalt content of field delivered Inconel 600 tubes is approximately 0.07-w/o cobalt. Within any one steam generator, the same statistical distribution of cobalt content of tubes exists. The manufacturers of Inconel have indicated that Inconel 600 with a guaranteed cobalt content less than 0.1-w/o for all material is unavailable at any price in the quantity needed for steam generator tubes.

b. Materials Availability

The material prices for various cobalt levels are based on the present availability of electrolytic nickel. If the supply situation of virgin nickel were to deteriorate and suppliers began using scrap to a greater extent, it is conceivable that the 0.10-w/o cobalt stainless could be unobtainable. For example, a sharp rise was noted in cobalt levels in unspecified stainless steel in 1973 due to the unobtainability of virgin nickel. The availability of low cobalt stainless also depends on the availability of low cobalt ferro-chrome. Since most of the United States' supply of chromium is from Third World countries, its availability cannot be ensured (nor can the prices).

c. Impact Evaluation

An impact evaluation, pertaining to reduction in cobalt content of materials, cannot be performed without taking into consideration alternate means of reducing occupational exposure. The basic philosophy is that dollars should be used where they can be most effective in overall product quality. The further reduction of cobalt content alone may not be cost effective in reducing radiation exposure. One must determine if there are other methods of reducing exposure at a lesser incremental cost per unit of exposure avoided.

Included in this impact evaluation is the need to consider the specific source of the cobalt that results in radiation exposures. As a first approximation, the radiation contribution might be equated on the basis of the surface area of the components in contact with the reactor coolant, weighted by the cobalt content of that material. The weights and surface areas are given in Table 12.3-2. On a weighted surface area basis (the product of the surface area and the cobalt content limit), it can be seen that the steam generators are predicted to be the largest contributor, as given in Table 12.3-5. In this type of analysis, it might be concluded that the cobalt content of materials should be limited in the steam generators.

The second approximation considers the corrosion rate of the stainless steel and Inconel in addition to the surface area and cobalt content. Using an approximation to the corrosion rate at steady state of 0.5 mg/dm^2-mo for stainless steel and 1.5 mg/dm^2-mo for Inconel 600 [6], the second approximation as given in Table 12.3-5 can be constructed. As shown in the table, the steam generator Inconel is predicted to be the primary contributor to the total cobalt inventory in the plant. This conclusion is further substantiated by comparing the iron to nickel ratio in deposited crud on the core. If the stainless steel has a corrosion release elemental distribution of Ni/Fe = 0.1, and assuming the Inconel has a Ni/Fe ratio = 0.5 (based on releasing Nickel ferrite, $NiFe_2O_4$), then the mixed corrosion products should have a ratio of nearly 0.5 for the corrosion rates given previously. Figure 12.3-3 gives a compilation of Ni/Fe ratios measured at operating plants. From the figure, it can be concluded that the mean Ni/Fe ratio is about 0.35. This is slightly less than that assumed for Inconel corrosion, but nearer to that assumed for Inconel than stainless steel. If the Inconel 600 corrosion is the major source of cobalt, then the percent of cobalt in the crud should be near that for the Inconel base metal.

Figure 12.3-2 shows the measurements of cobalt in core crud deposits from many plants over a number of years. The figure shows that most deposits have a cobalt concentration (in weight percent) less than 0.01 indicating that most of the crud comes from corrosion of materials very low in cobalt content. Figure 12.3-2 also substantiates the claim that Stellite is not a major contributor to the radiation levels due to Co-60 since the cobalt level in core crud deposits is very low.

Therefore, it is concluded that the present equipment specifications on cobalt content of materials wetted by the RCS are adequate.

However, implementation of a CPNPP cobalt reduction program provides additional assurance that efforts to limit and reduce cobalt based source materials are practiced especially for valve replacement and maintenance activities involving valves which communicate with the reactor coolant system.

4. Chemistry Control to Minimize Radiation Levels and Corrosion

a. Radiation Level Minimization

A third area of involvement in attempting to reduce the source of occupational exposure is in methods to reduce radiation levels in the reactor cavity refueling water. Upon transition of the reactor coolant from a reducing environment to an oxidizing environment during the refueling shutdown, a significant quantity of Cobalt-58 (100 to 2000 Ci) and a lesser quantity of Cobalt-60 is solubilized in the reactor coolant. The concept of peroxide (H_2O_2) addition to the reactor coolant after cooldown to 140°F has been considered in an attempt to effect the solubilization under controlled conditions. By the addition of peroxide, the solubilization occurs at such a time as the reactor coolant purification system can cleanup the circulating cobalt. The result of not adding peroxide at this time may be the solubilization when the reactor head is lifted, thereby allowing for mixing of the cobalt in the refueling water. It should be noted that the peroxide addition does not lower radiation exposures due to deposited crud nor is it a decontamination technique. It only allows for the solubilization of some radioactive crud (which would eventually solubilize anyway) under optimum conditions.

b. Corrosion Minimization

The hot hydro (hot) functional is conducted with only LiOH (no boric acid), which according to Riess [2] puts a protective coating on materials wetted by the RCS, thereby reducing corrosion.

Other programs such as the all-volatile steam generator chemistry might be considered exposure reduction programs as they are also aimed at reducing the maintenance, repairs, and inspection of aimed at reducing the maintenance, repairs, and inspection of steam generators.

The various decontaminations at Canadian National Deuterium Uranium reactors (CANDUs), boiling water reactors (BWRs), PWRs, and experimental reactors [7,8]

do not demonstrate that major decontaminations are either practicable or necessary at a PWR at this time.

While major decontaminations have reduced occupational exposures in the first few years after the decontamination, there is no evidence to show that long-term (e.g., greater than three to five years) exposures are reduced significantly in comparison to that expected with no decontamination. Surveys show that levels increase for the first two or three years after which a leveling trend is noted. Thus a system decontamination would reduce radiation levels only during this short time period. In evaluating the practicality of decontamination in PWRs, the extra radiation exposure accrued during the decontamination process must be compared to the projected savings in radiation exposure over the succeeding two year period. After this two year period, no savings should be projected due to the similarity of radiation levels thereafter and the large variability in radiation exposures from year to year. The overall practicality of decontamination must also include such factors as cost of the decontamination, cost of replacement power due to the extended shutdown period required for decontamination, and additional waste disposal cost. Further, decontaminations of CANDUs and BWRs do not show that decontaminations of large scale PWRs are practicable. Those techniques (including on-line decontamination) [8] used at CANDU reactors are not directly applicable to the PWR, since the reactor coolant chemistries are different.

CAN DECON (CANDU on-line decontamination) techniques cannot be used on-line but could be used four days after shutdown since the decon solution decomposes in a high radiation field. Since volume of coolant in an active core region is small in a CANDU, they get very slow decomposition. Since the volume of water in a core region is large (approximately 20-percent of RCS volume), in a PWR, it would decompose rapidly (approximately 15 minutes). At four days after shutdown, the decontamination solution would be effective for approximately 10 hours. CANDU experience indicates most of the decontamination is accomplished in approximately one hour with a DF = 6 for carbon steel and DF = 2 for stainless steel [9].

12.3.1.2.7 Criteria for Routing Radioactive Piping

There are no unshielded radioactive or potentially radioactive pipes in Zone I areas. (See Subsection 12.3.1.3.)

Pipes containing N-16 are routed behind the primary or secondary shield, or behind special shielding to a point where at least one min decay time has elapsed. Radioactive piping with contact dose rates in excess of the maximum design dose rate for a specified zone are not routed unshielded through areas where access is required for reading and adjustment of instruments, inspection, or maintenance and operation of nonradioactive equipment. Pipe shielding is designed to conform to the appropriate adjacent zone designation limitations.

12.3.1.2.8 Equipment Shielding

Equipment for radioactive systems is arranged in accordance with guidelines provided in the description of plant zoning given in Subsection 12.3.1.3. Major components such as tanks,

demineralizers, filters, and standard equipment packages are located in individual, shielded cubicles. Such equipment cubicles contain the minimum necessary piping so that access to these compartments is not needed during equipment operation. In the case of standard equipment packages where valves are not separated from the equipment, operation of these valves is performed from operating panels located in low radiation areas. The connecting piping and valves of small service pumps and major equipment are located in separate, shielded valve rooms. The lengths of pipes in these rooms are minimized to practical limits.

Inspection and maintenance of service pumps and valves can be accomplished when the system is not in operation. Manually operated valves in radioactive systems are operated from shielded valve operating areas located adjacent to valve rooms by the use of extension stems which penetrate the shield walls or slabs. Components of radioactive systems are fully shielded so that general areas such as corridors and stairways can be occupied by radiation workers 40 hr/wk, 50 wk/yr.

The standard equipment packages where valves are not separated from the equipment skid are the waste and recycle evaporators, the waste gas catalytic hydrogen recombiners, and the waste gas compressors.

The equipment packages are designed so that most valves are remotely manipulated. Locally operated valves on the equipment packages do not require manipulation during normal operation.

Valves on other components requiring durable packing are not used in the Westinghouse-supplied standard equipment packages other than the hydrogen recombiner located in the auxiliary building. Westinghouse provides diaphragm valves, control valves with bellows seals, or other valves which do not require durable packings. Graphite foil is used in the hydrogen recombiner, but since the recombiner is a gas processing equipment package it is not contaminated with corrosion products. Thus, maintenance of the recombiner valves does not constitute a major exposure source.

Furthermore, it is expected that the catalytic recombiner is a low-maintenance equipment package.

In addition, the equipment on the Westinghouse evaporator package skids are arranged so that serviceable equipment is separated from those components which accumulate significant amounts of radioactivity. Specifically, the arrangement is such that a 24-inch concrete wall separates the pumps, valves, and instruments from the evaporator concentrator tank, the gas stripper, and the vent condenser. Figure 12.3-13 illustrates this shielding wall in the recycle and waste evaporator cubicles.

Permanent flushing connections are provided on the waste evaporator, recycle evaporator, and catalytic hydrogen recombiner equipment packages. Flushing with water prior to maintenance assists in reducing exposure during maintenance on these units.

A sampling room for each unit is provided for sampling the primary plant systems. Personnel exposure to radioactive fluids is minimized by the use of sample hoods, isolation valves, sample vessel connections, and control of the sample pressure (by proper valve lineup) during all operations. Personnel exposure to radioactive gases is minimized by maintaining airflow through sample hoods, proper valve and sample vessel connection maintenance, and proper valve

operation. Permanent shielding is provided, where practicable, between the operator and the potentially highly contaminated components.

These equipment and facility design features limit occupational radiation exposure to ALARA levels. The equipment arrangements are provided in Figures 12.3-5 to 12.3-23.8.

12.3.1.2.9 Hot Lab and Counting Room

A Chemistry Hot Lab and Counting Room are provided to facilitate the handling, preparation, and analysis of radiochemical samples. The Hot Lab is located to provide easy access to the Primary Sample Rooms and is equipped with a fume hood that exhausts to the Primary Plant Ventilation System. Sink and floor drains in the Hot Lab are directed to the Liquid Waste Processing System. This room is equipped with an alarming area radiation monitor to detect and provide warning of abnormal radiological conditions. Additionally, remote readouts of area radiation monitors located in the Primary Sample Rooms are provided in the Hot Lab.

The Chemistry Counting Room is located adjacent to the Hot Lab and is designed to provide a low background area for performing radiological analyses of chemistry samples. A sample pass through window is provided between the Counting Room and Hot Lab to aid in contamination control and to minimize the handling of samples during transport.

12.3.1.3 Design Objectives and Zoning Guidelines

The facility, equipment, and radiation shielding are designed to provide protection against radiation for personnel and equipment, both inside and outside the plant, and for the general public. The plant radiation shielding is designed to perform the following functions:

1. Keep the radiation doses to plant personnel, construction workers, vendors, and authorized visitors within the limits set forth in 10 CFR Part 20.

2. Prevent excessive neutron activation and limit the accumulated dose to system components to specified limits.

3. Ensure that the dose rate levels in controlled access areas are kept below the maximum dose rate limit of the assigned zone.

4. Reduce radiation dose rates during inspection and maintenance caused by contained sources without unwarranted plant outages.

5. Permit access to and occupancy of the Control Room and Technical Support Center under accident conditions, including loss-of-coolant accidents (LOCA), and to limit the whole body dose to five rem for the duration of the accident as required by criterion 19 of 10 CFR Part 50, Appendix A.

6. Limit the dose rate at the exclusion area boundary caused by contained sources in the unlikely event of a LOCA.

The design of the facility and radiation shielding is greatly influenced by the radiation zoning philosophy used in the development of plant arrangements. Proper zoning of a plant facilitates

the arrangement of radioactive equipment in accordance with the requirements of 10 CFR Part 20 and the guidelines of NRC Regulatory Guide 8.8.

The following design guidelines are used to subdivide the plant into controlled and uncontrolled access areas as shown and described on Figures 12.3-4 to 12.3-23.8: For design purposes, the plant is subdivided into two areas: unrestricted access and restricted access. Unrestricted access areas will be occupied by personnel not involved in radiation work (Zone I).

The restricted access areas for radiation workers are subdivided into four zones. The first two zones are designed for personnel to perform routine work, Zone II for unlimited access time within a normal work week, and Zone III for limited periodic occupancy. The second two zones are designed for locating radioactive equipment and piping. In Zone IV, the amount of radioactive sources are limited to a practical minimum to allow access when needed. In Zone V, the amount of radioactive sources is not limited, so this designation is used for equipment and piping areas to which access is not normally required.

It should be noted that these radiation zone criteria are used for facility design and location of equipment and components based on expected maximum radiation levels; they are design assessments made with the provisions of 10 CFR 20.1-20.601. They are not intended for use as criteria for establishing requirements for radiation posting and access control during plant operations. Such operational requirements will be established in accordance with the requirements of the revised 10 CFR Part 20 based on actual radiation levels as discussed in Sections 12.5.3.2 and 12.5.3.3.

Zone I - Zone I areas are designed for no restriction on occupancy. Zone I areas are designed to limit radiation exposure caused by occupancy on a 40 hr/wk and a 50 wk/yr basis to the whole body dose limit of 0.5 rem/yr. Areas such as the Control Room, portions of the Turbine Building and offices are designated as Zone I areas. The maximum allowable external dose rate is 0.25 mrem/hr and represents a conservative estimate of the maximum values in areas within the station complex where plant personnel, construction workers, or routine visitors are permitted with unrestricted access. This upper limit is based on normal operation with one percent failed fuel and includes the effects of anticipated operational occurrences. Experience at operating pressurized water reactors, however, indicates that the average fraction of failed fuel is actually much less than 0.25 percent. A value of 0.12 percent is considered representative of expected operation. Accordingly, Zone I average external dose rates are expected to be well below the maximum. It is further expected that construction workers or visitors to the site will receive considerably less than 0.5 rem/yr limit because of the relatively short time they will be on the site.

Zone II - Zone II comprises the lowest of four levels of controlled access areas. Zone II areas are designed such that station personnel and authorized visitors can occupy these areas on a 40 hr/wk, 50 wk/yr basis and not exceed the allowable whole body dose of 1 1/4 rems per calendar quarter.

Corridors and areas outside radioactive enclosures where personnel can walk freely are included in this zone. Auxiliary equipment such as valves, dampers, and instrumentation and controls located in this zone can be maintained and operated without draining any equipment.

Average external dose rates and cumulative doses to individuals in Zone II areas are expected to be well below the maximum value cited in Table 12.3-6.

Zone III - Zone III constitutes the next level of areas designed for controlled access and includes areas where only weak radiation sources are expected to be present. Design of Zone III areas allows for personnel occupancy on a periodic basis consistent with cumulative exposure limitations. This zone includes valve-operating areas with stems which extend through shielding decks and auxiliary equipment such as valves, instrumentation, and controls.

Zone IV - Zone IV areas are designed to include high radiation areas with very restricted, controlled access and limited occupancy. Design of these areas considers the need for physical access controls such as lockable doors or gates or other physical barriers as discussed in 10 CFR Part 20.1601.

This zone includes all areas containing radioactive pipes, valves, pumps, heat exchangers, and tanks which need short-time access for operational activities such as reading gauges or manual operation.

Zone V - Zone V areas are designed to include those high radiation areas (10 CFR Part 20.1601) which are normally inaccessible when the reactor is at power or when equipment has large radioactive inventories. Design of these areas considers the need for physical access controls such as lockable doors or gates or other physical barriers as discussed in 10 CFR Part 20.1601.

Zone V is designated for large radioactive equipment which does not require maintenance while the reactor is at power, such as the primary coolant loop including pumps, steam generators, and the pressurizer.

Zone V is also designated for equipment such as the volume control tank, holdup tanks, gas decay tanks, demineralizers, filters, resin and waste storage tanks, pumps and heat exchangers which does not require maintenance when large radioactive inventories are present.

Table 12.3-6 presents a summary of the zoning criteria which was used to zone the facility. Figures 12.3-5 through 12.3-32 provide illustrative examples of equipment and piping layouts for liquid filters, demineralizers, recombiners, tanks, evaporators, pumps, steam generators, valve operating stations, and sampling stations. Instrumentation and control panel and motor control centers (MCCs) locations are provided on Figures 12.3-5 through 12.3-23.8. Sampling port locations are provided on Figures 12.3-24 through 12.3-32.

12.3.1.4 Decommissioning

The design features employed to maintain occupational radiation exposure as low as is reasonably achievable (ALARA) during operation of the plant are also applicable to decommissioning. These radiation protection facility design features are noted in:

Section 12.3.1.1 - Equipment Design Feature

Section 12.3.1.2 - Facility Design Considerations

Section 12.3.1.3 - Design Objectives and Zoning Guidelines

These sections include such features as equipment arrangement for ease of accessibility and maintenance; component design to minimize crud buildup; provisions for decontamination, temporary shielding, and development/use of remote handling equipment; provisions for remote

flushing of equipment; and specifications and limitations on the cobalt content in equipment components to minimize the buildup of activated corrosion products. The specifications and limitations on cobalt content in equipment components will serve to limit radiation doses from crud buildup during both operation and subsequent decommissioning. A summary of the features in Westinghouse PWRs that reduce occupational exposure are given in Reference [13]. Further discussion of design considerations relevant to maintaining personnel exposure ALARA during decommissioning are provided in Section 12.1.2.

With these types of features, the AIF study on decommissioning [12] has estimated occupational radiation exposure for PWRs for the primary decommissioning alternatives as follows:

Mothballing	150 man-rem
Entombing	130 man-rem
Prompt Removal/Dismantling	630 man-rem
Mothballing - Delayed Removal/Dismantling	460 man-rem
Entombing - Delayed Removal/Dismantling	440 man-rem

12.3.2 SHIELDING

12.3.2.1 Shield Design Criteria

In general, radiation shielding throughout the plant is designed to meet the specifications established in 10 CFR Part 20 and to keep personnel radiation exposure ALARA. More specifically, the criteria for penetrations presented in Subsection 12.3.1.2.2, the zoning criteria, and the general guidelines stated in Subsection 12.3.1.3 are used as a basis for shielding calculations and engineering decisions affecting potential radiation exposures.

12.3.2.2 Calculation Method

Shield wall and slab thicknesses are determined by using basic shielding data and equations. Nuclear data is taken from Nuclear Data Sheets, National Academy of Sciences [11]. The methods used for calculations are described in The Engineering Compendium of Radiation Shielding [10].

Computer codes used for the original shielding design are as follows:

1. COSACS - Program calculates and tabulates dose rates from cylindrical radiation sources through finite thick slab shields using the point kernal integration method. It also plots results (dose rate vs. shield thickness).

2. ACTIV 4 & 5 - These programs calculate the source term for composite gamma radioactive sources (MeV/cm^3 sec) from nuclide activity inventories (Ci/gm) into source strength composition based on selected energies.

3. HOMES - Program converts heterogeneous equipment sources, consisting of a cylindrical vessel with interior tube bundles into a homogeneous cylindrical volume source.

4. CREED - Program computes environmental and Control Room doses from the radiation sources at given distances and for various wall and roof slab density by means of (a) Geller-Epstein's Source Integral, (b) source supplied by manufacturers, (c) already calculated source.

5. G^3 (G cubed) - Program computes fluence or dose rates from a point radiation source for direct and single scattered gamma radiation.

The geometry used for shielding evaluation is that of a finite shielded cylinder for tanks, demineralizers, filters, heat exchangers, and pipes.

Calculations are performed to determine the gamma dose rate through a laminated shield at the sides and ends of a cylindrical source to obtain exposure values. Source strengths are divided into selected energy bins for different gamma energies. Credit is taken for self-attenuation.

12.3.2.3 Shielding Material

The bulk of shielding material is ordinary concrete with a minimum density of 141 lb/ft^3 after 28 days of curing. Commercial non shrink grout and mortar having a minimum density of 130 lb/ft^3 may be utilized in congested areas when closing temporary construction blockouts. In the Containment wall, however, concrete with a minimum density of 136 lb/ft^3 is used. The amount of reinforcement per cubic foot of ordinary concrete shield walls and slabs is a minimum of 6.26 lb. Where space requirements prohibit the use of ordinary concrete, high density concrete (220 lb/ft^3) with ilmenite and hematite aggregate is used. Observation windows in the drumming area and filter transfer area are of high-density glass with attenuation properties that provide adequate shielding to reduce radiation levels to be within specified dose limits (zones).

12.3.2.4 Shield Descriptions

12.3.2.4.1 Primary Shield

The primary shield is a large mass of reinforced concrete, which is a minimum of six feet thick throughout the full-length of the active reactor core and of irregular geometrical configuration. It surrounds the reactor vessel and extends upward from the Containment mat to form the walls of the refueling cavity. The primary shield is designed to perform the following services:

1. Reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel and allow limited access to the Containment during normal full power operation.

2. Limit the radiation level after shutdown from sources within the reactor vessel and permit limited access to the RCS equipment.

3. Limit neutron flux activation of component and structural materials over the life of the plant.

12.3.2.4.2 Secondary Shielding

The secondary shield is a reinforced concrete wall and slab structure that surrounds the RCS equipment, including pipes, pumps and steam generators. This shield, which is a minimum of 2 ft 9 in. thick, protects personnel from direct gamma radiation emanating from reactor coolant fission and activation products that are carried from the core by the reactor coolant. The secondary shield reduces the radiation from the reactor and doses from the N-16 activity in the primary loops. It supplements the primary shield to attenuate radiation to low levels; this permits limited access to the Containment for inspection during full-power operation and maintenance of equipment in certain areas during hot shutdown.

The secondary shield provides protection for refueling operations, inspection, repair and maintenance during refueling, and shutdown periods. During the transport of spent fuel assemblies, concrete shielding provides protection in the spaces close to the fuel transfer route and the reactor internals temporary storage area. The fuel transfer route is flooded with water which protects the airspace above the fuel transfer path. For a more detailed discussion of fuel transfer tube shielding, see Section 12.3.2.4.7 below.

12.3.2.4.3 Containment Building

The Containment Building is a cylindrical reinforced concrete structure that completely surrounds the NSSS. At full-power operation, this shield attenuates the radiation level outside the primary-secondary shield complex to ensure that radiation levels outside the Containment correspond to zone designations of adjacent areas. The Containment wall and dome are 4-1/2 ft and 2-1/2 ft thick, respectively. These thicknesses ensure that the cumulative dose from sources contained within the plant during the 2-hr period following a DBA would not exceed 10 CFR Part 100 guidelines.

12.3.2.4.4 Auxiliary Shielding

The auxiliary shielding consists of wall and slab structures for various systems which carry process fluids and gases or handle radioactive materials. Certain equipment located in the Safeguards Building are assumed to be radioactive during normal operation, or could become so under special conditions. Such components are analyzed to establish their maximum radiation levels and the shielding is sized accordingly. Equipment compartments are individually shielded from radioactive components in adjacent compartments.

Auxiliary shielding includes all concrete walls, slabs, covers, and removable blocks that shield the numerous radiation sources in systems such as the CVCS, the BRS, and the RHR system. Highly radioactive equipment is isolated individually, or in groups, in shielded compartments. The entrances of such compartments are provided with labyrinths for full protection of adjacent areas. Where concentrations of radioactive equipment and piping occur, the associated manually operated valves are grouped in adjacent valve rooms. Pumps and compressors serving highly contaminated tanks are located in separate shielded compartments.

Auxiliary shielding thicknesses are designed so that dose rates in adjacent areas are below the limitations set forth by the zone designations of such areas.

12.3.2.4.5 Refueling Cavity

The refueling cavity is flooded with water during refueling operations. In addition to providing laydown space for the reactor internals, the cavity serves as the transfer medium in which spent fuel is moved from the reactor to the spent fuel area. Portions of the cavity are sized on the basis of the internals package physical size and moving path; the remainder is sized on the basis of the spent fuel physical size and moving path.

Shielding provides protection during all phases of spent fuel removal and storage. Operations that require shielding of personnel include spent fuel removal from the reactor, temporary spent fuel storage (in the Containment), spent fuel transfer through the refueling canal and transfer tube, spent fuel storage (in the Fuel Building), and spent fuel shipping cask loading prior to offsite transportation. All spent fuel removal and transfer operations are performed under borated water to provide radiation protection.

During movement of fuel assemblies within containment, at least 23 feet of water is maintained over the top of the reactor vessel flange. At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks and 10 feet in the transfer canal. This limits the dose at the water surface to less than 2.5 mrem/hr for an assembly in a vertical position.

The 6-ft-thick walls of the fuel transfer canal and 6-ft-thick spent fuel pool walls supplement the water shielding and limit the maximum radiation dose levels in adjacent working areas to less than those specified by their zone designations.

The refueling water and concrete walls also shield personnel from activated control rod clusters and reactor internals that will be removed during refueling. Dose rates generally are expected to be less than 2.5 mrem/hr in working areas. However, certain manipulations of fuel assemblies, rod clusters, or reactor internals may produce short term exposures in excess of 2.5 mrem/hr.

12.3.2.4.6 Control Room Shielding

The Control Room shielding is described in Section 6.4.2.5

12.3.2.4.7 Fuel Transfer Tube Shielding

All the unrestricted areas are shielded from the fuel transfer tube either by stationary concrete or permanently installed removable shielding or by water shielding during spent fuel transfer. Access to the fuel transfer tube is further restricted by removing the inspection access ladder.

Figures 12.3-33 sheets 1 through 3 show the shielding design on the relevant plan, elevation, and section drawings, to assure acceptable levels of radiation in potentially occupied areas. Special attention is given to gaps in the concrete and lead shielding which are required for structural reasons.

Dose rate at contact with the 8" lead shielding is less than 6 Rads/hr where the direct radiation path is through the 8" lead shield. It is provided to shield as much of the 4" structural gap as possible. It supplements the existing permanent shielding concrete structures and water as shown in Figure 12.3-33 (3 sheets). It provides radiation protection for structural gap areas above and below the transfer tube by reducing the streaming and scattering of gamma rays.

The large gap (approx. 18") in the lead shield shown in section 3-3 of Figure 12.3-33 (sheet 3 of 3) is to provide more access space near the fuel transfer tube area in the inspection shaftway. This shaftway extends from elevation 808' up to the fuel transfer tube area, and is also shown in Figure 12.3-33 (sheet 1 of 3).

Within the Fuel Building and the Containment Building, the provision of shielding material above the transfer tube in conjunction with administrative procedures provides adequate protection to assure compliance with 10 CFR 20.

To further reduce dose rates from scattered radiation during spent fuel transfer, additional steel shield plates are installed at strategic locations around the fuel transfer tube (see Fig. 12.3-33, sheet 3 of 3). The shielding provided by these steel plates in conjunction with the lead shielding and administrative procedures reduces maximum expected dose rates in the vicinity of the fuel transfer tube area at 808’ to less than 1.0 rem/hr.

12.3.2.5 Use of Guidance Provided by Regulatory Guides

NRC Regulatory Guide 1.69 states that ANSI N101.6-1972, Concrete Radiation Shields, is considered applicable to shielding structures for nuclear power plants, subject to certain conditions. These conditions are stated in regulatory positions Nos. 1 through 8 in NRC Regulatory Guide 1.69. The guidance provided in this regulatory guide is adhered to as discussed in Appendix 1A(B).

12.3.3 VENTILATION

12.3.3.1 Design Objectives

The plant ventilation systems are designed to provide suitable and safe ambient conditions for personnel and operating equipment. In addition to the conventional function of preventing extreme thermal environmental conditions, the ventilation systems are designed to ensure that the maximum airborne radioactivity levels for normal and anticipated operational occurrences are within the limits of 10 CFR Part 20, Appendix B, for areas within the plant structures and for restricted areas on the plantsite where workers and visitors are permitted. The maximum levels correspond to the design basis reactor coolant isotopic inventory. The plant ventilation systems are also designed to provide an environmentally suitable and radiologically safe environment for continuous personnel occupancy in the Control Room under normal and postaccident conditions in accordance with 10 CFR Part 50, Appendix A, Criterion 19, and to ensure that the maximum radiological exposures to the general public following a DBA are within the limits of the guidelines of 10 CFR Part 100.

In order to meet the design objectives for the plant ventilation systems, the following design guidelines are incorporated into the design basis for the ventilation systems:

1. Air movement patterns are provided from clean areas to potentially contaminated areas, thus preventing the spread of airborne contamination.

2. Negative pressure differentials, with respect to surrounding areas, are maintained, where applicable, inside potentially contaminated areas to prevent uncontrolled spread of airborne contamination.

3. Atmosphere cleanup units are installed to filter the exhaust of potentially radioactive areas in order to maintain the offsite doses ALARA. These offsite doses result from the activity of the ventilation discharge.

12.3.3.2 System Design Criteria

The atmosphere cleanup units remove exhaust air contaminants from the primary plant ventilation exhaust system (which includes the controlled access exhaust, containment purge exhaust, and Fuel Building exhaust ventilation systems) and from the Hydrogen Purge System.

These systems are described in Sections 6.2, 6.5, and 9.4. Figures 6.5-1 and 9.4-10 illustrate typical layouts for the engineered safety features (ESF) and non-ESF atmosphere cleanup units.

In addition to the use of the atmosphere cleanup units referred to previously, the emergency filtration units and emergency pressurization units of the Control Room heating, ventilation, and air-conditioning (HVAC) system remove contaminants from the Control Room environment and prevent contaminants from entering the Control Room atmosphere during emergency modes of operation. The Control Room HVAC system is discussed in Sections 6.4 and 9.4.1. Atmosphere cleanup units are also being used in each Containment Building. These preaccess filtration units are described in Section 9.4.

12.3.3.3 Component Design Criteria

The normal primary plant ventilation exhaust system atmosphere cleanup units and the ESF Control Room emergency filtration units each contain a prefilter, two high-efficiency particulate air (HEPA) filters, a charcoal bed filter for iodine adsorption, and related instrumentation. The HEPA filters are located immediately upstream and downstream from the charcoal bed filter. In addition to the components previously described, the ESF Control Room emergency pressurization and Hydrogen Purge Exhaust System atmosphere cleanup units each contain a demister and electric heater section to ensure that the relative humidity of the incoming air is maintained below 70 percent. The Primary Plant ESF atmosphere cleanup units which normally remain on standby are similar to the Control Room emergency pressurization and Hydrogen Purge Exhaust System atmosphere cleanup units, except that they do not contain a prefilter section.

12.3.3.4 Testing, Isolation, and Decontamination

Each atmosphere cleanup unit housing is furnished with test connections for dioctyl phthalate (DOP) leak testing of the HEPA filters and refrigerant leak testing of the charcoal bed iodine adsorber section. In addition, a water spray system is provided to suppress an adsorber section fire caused by the presence of activated charcoal in the atmosphere cleanup units. The operation of the water spray system is discussed in Section 9.5. Inspection and testing requirements for ESF and non-ESF atmosphere cleanup units are discussed in Section 9.4. The arrangement of these units allows access to the unit and equipment by personnel for maintenance and testing. All housings have multiple drains which are connected to the Equipment and Floor Drainage System. Decontamination water stations are in proximity to each unit to facilitate decontamination. Figures 6.5-1 and 9.4-10 illustrate typical layouts for ESF and non-ESF atmosphere cleanup units, respectively.

12.3.3.5 Maintenance

All atmosphere cleanup unit housings are designed to provide ample room for maintenance of equipment and filters and to minimize radiation exposure to personnel during filter replacement. The criteria for replacement of filters are established on the basis of maximum allowable resistance of the dirty filter or minimum radiation exposure to personnel, or both. Each atmosphere cleanup unit is provided with a pressure indicator. High differential pressure across the entire unit actuates an alarm in the Control Room. Each filter section has provisions to monitor its own status with a local differential pressure indicator. Upon reaching the maximum resistance, the filters are replaced. The maximum allowable resistance for each type of filter is indicated in Table 9.4-4.

The criterion for replacement of the adsorber section is based on the deterioration of adsorber ability to remove radioactive iodine from the exhaust air. This efficiency is determined by laboratory testing of representative samples of the activated charcoal exposed simultaneously to the same service conditions as the adsorber section. Each sample has the same qualification and batch test characteristics as the system adsorber. Samples are tested periodically in accordance with Regulatory Position C.6.b of NRC Regulatory Guide 1.52 or 1.140. The adsorber section will be replaced if one of the samples fails to meet the requirements of NRC Regulatory Guides 1.52 or 1.140, Table 2. See Appendix 1A(B) for a discussion of compliance with NRC Regulatory Guide 1.52 and 1.140.

12.3.3.6 Conformance to NRC Regulatory Guide 1.52

The ESF atmosphere cleanup units conform to the regulatory positions and recommendations of NRC Regulatory Guide 1.52 as indicated in Table 6.5-1 and Appendix 1A(B). Non-ESF atmospheric cleanup units conform to the regulatory positions and recommendations of NRC Regulatory Guide 1.140 as indicated in Appendix 1A(B).

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Airborne Radioactivity Monitoring System

Detection and measurement of airborne radioactivity is accomplished by the process radiation monitoring system (PRMS) in conjunction with the area radiation monitoring system (ARMS). Together, the PRMS and ARMS comprise the CPNPP Radiation Monitoring System (RMS). PRMS and ARMS monitor activity levels and dose rates resulting from gaseous and particulate airborne radioactivity in various areas of the plant; the systems provide alarms, indication, and on-demand hard copy printouts in the Control Room.

When an alarm is annunciated in the Control Room, Radiation Protection (RP) will be contacted. RP personnel will establish access control to the identified building(s) to prevent inadvertent entry of personnel. RP personnel will then seek to further identify and locate the specific source or source location within the building(s) identified. This is accomplished by means of a review of area radiation monitor readouts for abnormal levels (utilizing the computer capabilities of the RMS as described in Section 11.5.2.5.3), collecting local air samples in specific areas (using mobile air sampling systems), and communication with Operations personnel to identify abnormal system configurations, component failures or other possible causes. A listing of all local monitoring assemblies and a discussion of their alarm setpoints is provided in

Section 11.5.2.5.6. For further discussion of the role of RP personnel, see Section 12.5. Detailed descriptions of PRMS/ARMS are given in Sections 11.5 and 12.3.4.2. Portable instrumentation and the airborne sampling program are discussed in detail in Sections 12.5.2 and 12.5.3, respectively.

12.3.4.1.1 Design Criteria

The process radiation monitoring system, which performs airborne radioactivity monitoring, is designed to measure, display and store airborne radioactivity levels, to alarm on high airborne radioactivity levels, and, when required, to control the release of radioactive gases and particulates produced in the operation of the plant. It also aids compliance with the requirements of 10 CFR Part 20, 10 CFR Part 50, General Design Criterion (GDC) 60, 63 and 64 and NRC Regulatory Guides 1.21 and 8.8. The process radiation monitoring system aids in the protection of the general public and plant personnel from exposure to airborne radioactivity in excess of that allowed by applicable regulations, by, when appropriate, controlling or terminating releases exceeding discharge limits and warning plant personnel so that they can take appropriate protective measures. The design objectives of the fixed system for normal operation are as follows:

1. To give early warning of increasing radioactivity levels indicative of equipment failure or malfunction or deteriorating system performance.

2. To initiate prompt corrective action, either automatically or through operator response, on high airborne radioactivity level.

3. To provide continuous surveillance of noble gas radioactivity levels within certain buildings by indicating and recording in the main Control Room, exhaust duct activity levels for up to 28 days. Abnormal activity levels are alarmed in the Control Room to aid in preventing a person from inadvertently entering a building where he can inhale noble gas activity in excess of limits defined in 10 CFR Part 20. Also, there are area monitors in these buildings which will alarm locally upon detection of radiation due to increased airborne radioactivity in these areas (see Section 12.3.4.2.2 for monitor location bases).

For some anticipated operational occurrences resulting from accidents or operator error, e.g., if the airborne radioactivity levels exceed alarm set points, the PRMS will activate necessary isolation valves to terminate or reduce releases. Independence of backup monitors is maintained by providing adequate separation of detectors, signal cabling, power supplies, and actuation circuits for isolation valves.

The fixed continuous monitors, which are described in detail in Section 11.5, serve in conjunction with a comprehensive air sampling program using portable airborne sampling equipment. The portable air sampling equipment and the program for performing air sampling are described in detail in Sections 12.5.2 and 12.5.3. The portable airborne sampling equipment is available to complement fixed instrumentation and maybe used in areas where fixed instrumentation is not provided but where there is a need for short term or periodic monitoring or when fixed instrumentation is not operable. Portable radiation and radioactivity monitoring equipment are highly reliable, commercial quality equipment. The operation, maintenance and calibration of this equipment is subject to the Operations Quality Assurance Program. This equipment either collects samples which are then processed and evaluated, or monitors continuously. The system

also performs post-accident functions as described in Section 6.2.4 and 7.5 to satisfy GDC-13 and 54.

12.3.4.1.2 Airborne Monitor Locations

Fixed airborne monitors of the process radiation monitoring system provide information about the airborne activity within the following plant areas (described in Section 11.5):

1. Containment Building

2. Auxiliary Building

3. Heating and ventilation equipment area

4. Fuel Building

5. Safeguards Building

6. Main steam and feedwater area

Locations of airborne monitors that monitor effluent streams and the above areas are listed in Table 11.5-1.

12.3.4.1.3 System Description

The airborne monitors are part of the process radiation monitoring system, which is a dual computer and distributed microprocessor digital monitoring system described in detail in Section 11.5.

Detection of airborne activity in the Containment buildings during accident conditions is provided by the Containment High Range Radiation Monitors (Section 12.3.4.2) located inside Containment, and by grab samples for laboratory analysis.

The Containment particulate, iodine and noble gas (PIG) monitors are located outside the Containment building in the Safeguards buildings at elevation 831 ft, 6 in. The sample collector draws containment air through an open sample line from the Containment to the monitors. A correction factor was determined to account for particulate deposition line losses. The sample pump for the containment PIG monitor is located with the monitor in the Safeguards Building. The guidelines in ANSI N13.1 are considered to ensure that representative samples are obtained.

Various areas of the plant that have potential airborne activity have their ventilation directed to a common header that discharges to the plant vent stacks. Each stack, located on the Auxiliary Building, has off-line noble gas monitors (see Section 11.5) with sample pumps monitoring plant effluents. Each stack is also equipped with an off-line Wide Range Gas Monitor (WRGM) used to monitor both normal and post-accident ranges of noble gases. The WRGM has sample pumps and isokinetic sample nozzles for both normal (high stack flow rate) and accident (low stack flow rate) conditions. Both types of monitors draw representative samples of stack effluent and route them to the detectors. A correction factor was determined to account for particulate deposition line losses.

There are also in-line gaseous monitors in the vent ducts of the Auxiliary, Fuel, and Safeguards building, the heating and ventilation room exhaust duct, and the main steam and feedwater area. They are located at the outlet of each area or building ventilation header. These monitors have no sample lines or pumps associated with them because they are located in each area's ventilation duct. The monitors are beta sensitive scintillation detectors with photomultiplier tubes and associated electronics. They provide continuous service (described in 12.3.4.1.1, Objective 3) and are not designed to detect particulates or iodines. The monitors identify which building has abnormal airborne activity and provide indication that some system within the building or area being ventilated has equipment leaking significant gaseous or liquid activity. Refer to Table 11.5-1 for monitor description and parameters.

The Control Room is provided with four gas monitors (2 per intake) each capable of monitoring Control Room ventilation inlet air. A high radiation signal will be initiated upon detection of sufficient airborne radioactivity in the inlet air and will initiate Control Room emergency ventilation and start recirculation of Control Room air through filters. The monitors are beta sensitive scintillation detectors with photomultiplier tubes and associated electronics. Two of the monitors are located near the Control Room ventilation duct in the heating and ventilation area of the Electrical and Control Building. The other two monitors are located in the Auxiliary Building.

Section 11.5 describes in detail the PRMS and the Digital Radiation Monitoring System (DRMS), monitor calibration methods and frequency, tests and inspections.

The off-line Containment PIG monitors are seismically designed monitors. They have digital indication on the control/display module installed on the seismic equipment rack, located in the Control Room. All airborne monitors have remote indication, alarm, and printout via the peripherals of the microcomputer consoles located in the Control Room.

12.3.4.2 Area Radiation Monitoring System

The objectives of the ARMS are as follows:

1. To provide plant operators and personnel with a system which informs them of radiological conditions is selected plant areas and provide an indication of changing radiological conditions.

2. To indicate, alarm, and record for up to 28 days abnormal radiation levels in areas where radioactive material is present, stored, handled, or inadvertently introduced.

The primary functions of the ARMS are as follows:

1. To provide continuous surveillance of radiation dose rates of representative accessible and restricted areas inside reactor containment.

2. To alert personnel of dose rates above a predetermined set point.

3. To provide a direct reading indication and on demand record of daily radiation dose rate averages for up to 28 days for each monitor location.

4. To indicate in the Control Room the radiation levels at selected locations within the plant building where spent fuel is stored or handled.

5. To warn operating personnel when a monitor is not operational

6. To have sufficient range and sensitivity to monitor normal and anticipated operational occurrences

7. To monitor the Containment radiation level after a DBA (see Section 7.5).

8. To monitor area radiation in locations where access may be required after a DBA (see Section 7.5).

9. To monitor area radiation in areas adjacent to containment which contain electrical or mechanical penetrations (see Section 7.5).

Locations of the area radiation monitors are shown on Figure 12.3-4 through 12.3-23.

12.3.4.2.1 Design Criteria

The following design criteria are used in the design of the ARMS:

1. Detectors and cabling for safety related monitors located in containment are designed to operate during and following the applicable DBA, including LOCA.

2. To facilitate compliance with applicable regulations (e.g., 10 CFR Part 20), monitors and detectors have matching ranges in accordance with dose rate levels anticipated at specific detector locations.

3. The system is designed to provide for physical security for access to alarm setpoints and other important data base parameters.

4. Radiation dose rates are continuously monitored, indicated and annunciated on the Control Room's Display/Operator Cabinets, and recorded on demand at the associated printers.

5. Local audible alarm and analog indication are provided for selected ARMS microprocessor or detector location.

6. Control Room alarms annunciate high radiation dose rates and circuit failures.

7. Monitors electronics have independent internal power supplies.

8. A remotely operable check source or current is provided for each detector.

9. Environmental design conditions for monitoring components are described in Table 12.3-9.

10. Low and high-range area monitors register full-scale if exposed to radiation levels up to 100 times full-scale indication; they do not fold over throughout this range.

11. Monitors are readily accessible for maintenance and inspection wherever possible.

12. The unique function of each monitor is considered in determining alarm set points.

The specified range for each area radiation monitor is provided in Table 12.3-8. The full scale value for each monitor is, as a minimum, in the upper half of the top decade. For example, an ionization chamber detector with a specified range of 10^2 -10^7mR/hr will have a full scale reading of at least 5x10^6mR/hr.

12.3.4.2.2 Monitor Location Bases

Area monitors are located in representative areas (See Table 12.3-8) where the following occurs:

1. Personnel perform infrequent duties, but a significant probability exists for hazardous dose rates (i.e., inside containment).

2. Accident monitoring requirements apply (see Section 7.5)

3. Nuclear material is handled, used, or stored (i.e., R.G. 1.13).

12.3.4.2.3 System Description

The ARMS is part of the microprocessor-based radiation monitoring system. This digital monitoring system is described in detail in Section 11.5. System parameters, nominal instrument range, alarm set point bases, and the location of detectors are described in Table 12.3-8. Location and type of readouts and alarms for the airborne monitor are described in Section 11.5. Location of local readouts and alarms for area monitors are shown on Figures 12.3-4 through 12.3-23.

The basic ARMS (integrated with PRMS) is comprised of four dedicated computers in communication with each other (two in each of two central display consoles), distributed dedicated microprocessors (one for each local detector/monitor assembly), and a report computer. At each monitor, control, data processing, data storage, and multilevel alarming are performed locally by the dedicated microprocessor; also, processed data are communicated to central consoles upon polling requests. These monitor functions are performed at each monitor independently of the rest of the system. This independence is insured by use of optical couplers in monitor input/output circuits and by the distances separating monitors. No area monitor electronics share a microprocessor or internal power supply. Thus, each monitor is a stand-alone system performing independently and locally all functions of a conventional analog ratemeter, but, in addition, providing data storage and self-contained function-checking diagnostic routines in read only memory (ROM).

In addition to a 120-V, single-phase, 60-Hz local power supply for each monitor assembly, field wiring consists of two shielded, twisted-pair cable daisy-chained from monitor to monitor and to display consoles. Each daisy chain is a bidirectional control and data bus looped from one central computer through a group of monitors to the other central computer, operating at logic voltage and power levels. With this arrangement, two paths from each monitor to the central consoles are provided. See Figure 11.5-1.

The central Display/Operator Cabinet includes two microcomputers, an 18-in. color LCD display, a shared color graphics printer, printers, and keyboards for entering data and commands. The

cabinet is located in the common control room area for Units 1 and 2. The cabinet is divided into two sections, each section can display data associated with any monitor or monitor group in the plant.

Each operator console view node includes an 18-in. color LCD display and a keyboard for entering commands. One view node is located at the operator's console (CP1-ECBDCB-21, Operator Console B) in the Unit 1 control room area and the other is located at the operator's console (CP2-ECBDCB-21, Operator Console A) in the Unit 2 control room area. Each view node can display data associated with any monitor or monitor group in the plant.

1. Detectors

   The following two types of ARMS detectors are used: Geiger-Muller tubes for lower-level radiation and ionization chambers for the higher levels. (See Table 12.3-8.) There is an indicator and visual/audible high alarms at selected local microprocessor locations. In addition, certain detectors located remotely from the associated microprocessor are also provided with meters and alarms.

2. Local Electronics

   Associated with the detector is an Intel 8085A microprocessor with sufficient memory to control the monitor, process and store its data, and communicate processed results to the two, identical remote processors in central consoles.

   The monitor microprocessor and all of the generalized digital circuits are included on a board that is designed to fit into a National Electrical Manufacturers Association (NEMA) 12 enclosure with a lock. A printed-circuit backplane board accepts the processor board and the digital preamp, relays, power supply, and an address plug. The address plug gives the monitor a unique address. Also included are the detector high-voltage power supply, analog indicator, and indicator lights.

3. Display/Operator Cabinet

   The RMS Display/Operator Cabinet is a common panel divided into two (2) sections (SCADA A an B). Each section includes two IBM industrial personnel computers with math coprocessor. One computer is a SCADA node and contains a realtime interface coprocessor which allows it to communicate with the local microprocessor of each monitor. The other computer is the operator console view node. An IBM multistation access unit (MAU) interconnects the two computers to an IBM token-ring network (TRN). Each computer stores in memory the complete alarm status for all monitors in the Unit 1 and Unit 2 ARMS. Each section also includes a color LCD, an output printer, a shared graphics printer and an input keyboard. Each operator console view node includes a display color LCD Display and an input keyboard. Selection of the proper keys on the console keyboards can display the complete alarm status on the color LCD Display via a grid presentation and a control menu display, with each monitor represented by a colored rectangle with six superimposed characters. The characters identify the monitor. The color of the rectangle indicates the urgency of the alarm condition. Similar information is provided in Section 11.5.

4. Power Supplies

The radiation monitor processors and the central Display/Operator Cabinet are furnished with safety related or reliable non-safety related power (backed by diesel/non-safety related station batteries) as applicable.

Operating voltages for local electronics and high voltage for detectors are derived from power supplies in their respective local monitor-assembly enclosures.

5. Calibrations and Source Checks

A channel calibration is performed for each safety related area monitor channel at least every 18 months or during refueling outages. The channel calibration for non-safety related area monitors is performed at a frequency determined by plant procedure(s), based on performance and function. These calibrations are performed in accordance with approved station procedures. Electronic calibrations and checks, including alarm initiating actions, for each area monitor channel are performed in the field. Calibration of all area monitor detectors, with the exception of the post accident High Range Radiation Monitors located in containment, is performed as follows. Prior to initial installation in the field, each area monitor detector will receive a primary calibration onsite in the Radiation Protection Calibration Facility using a calibration source to generate radiation fields and a test microprocessor to test detector response. The onsite primary calibration process consists of determining the proper operating voltage for each detector and then testing the detector in radiation fields of at least three dose rate levels to determine detector response and demonstrate detector linearity. Traceability of the calibration to the National Bureau of Standards (NBS) is obtained by the use of a transfer instrument with an NBS traceable calibration to quantify the radiation fields generated by the calibration source. Upon completion of this calibration process, the detector is exposed to a portable transfer source and a reference detector response is determined. The detector is then installed in the field and the same transfer source is used to verify proper detector response. After initial installation in the field, subsequent calibrations of the detector are performed by either removing the detector from the field and repeating the primary calibration procedure described above, or by performing a transfer calibration in the field using the portable transfer source and reference data generated in the original primary calibration.

Calibration of the post accident High Range Radiation Monitors located in containment is performed in place using a portable source to verify detector response at a single point below 10R/hr. The monitor performance for points above 10R/hr is verified electronically by simulating detector input with an electronic signal.

Each low range area monitor GM detector assembly is provided with a built-in radiation check source. The check source is attached to the detector assembly in a shielded container in a fashion such that loss of check source actuation power will cause the source to return to a shielded position. Higher range area monitor ionization detectors have ranges that are too high to check with radiation sources. These are provided with check currents that test the electronic circuitry of the monitor. These monitor checking capabilities are remotely operable from the control room. These checks provide convenient operational checks of these monitors. Additionally, the radiation check source provides a gross response check of the detector.

System operability may be verified by observation of channel behavior or by use of the check source or current. Any detector whose response cannot be verified by these methods may have its response checked with a portable check source.

6. Test Calibration Circuitry

Each monitor microprocessor contains self-checking diagnostics in its ROM, a rotary switch on a printed circuit (PC) card to select each diagnostic program, and a red and green light (also on each PC card) to show the result of each diagnostic test. These allow the microprocessor board to be functionally checked while installed in its enclosure. Typical tests include the following:

a. Communication loop (continuity)

b. ROM

c. Local indicator

d. Local alarms

7. Maintenance

Area monitor channel detectors and electronics are serviced and maintained to ensure reliable operations. Manufacturer's recommendations for Service and Maintenance are available for use. Such maintenance includes cleaning, adjustment, and/or replacement of any components required after performing a test or calibration. If any work is performed that could affect the calibration, a recalibration will be performed at the completion of the work.

8. Control Room Equipment Rack#78#

A dedicated control/display module for each high range radiation monitor (two per containment building) is located within a Class 1E, seismic equipment rack in the Control Room. This equipment rack has the same characteristics and provides the same monitor indications as the control room equipment rack for selected process monitor channels (described in Section 11.5.2.5.5), except that power to the rack is provided directly by class 1E buses.

12.3.4.3 Accuracy

The overall accuracy of a radiation monitor system is governed by four major areas: (a) factory calibration and alignment, (b) detector characteristics and environment, (c) microprocessor environment, and (d) field alignment. Factory calibration is performed with standards traceable to the National Bureau of Standards (NBS). Detector energy response and linearity are demonstrated by factory calibration and site calibration procedures subsequent to operations. Detector and microprocessor environmental variations are determined from qualification tests. Overall monitor error is the root-mean-square sum of all system errors and for all conditions is within a factor of 2 over the entire range. An exception to the above accuracy statement occurs with the High Range Radiation Monitors (HRRMs). With a post Main Steam Line Break (MSLB) high temperature environment, a drop in the insulation resistance of the signal cable could cause

an erroneous bias current that could exceed three times the threshold radiation current. This situation, combined with other possible system inaccuracies, could exceed the stated accuracy limits. This will not affect the intended use of the monitor.

Another exception to the accuracy requirements has been identified with the High Range Area Monitors (HRAMS), at the low end of the monitors scale. When the dose rate from the keep alive source is combined with other system errors the indicated dose rate can exceed the previously specified system accuracy. This inaccuracy is conservative.

The precision of a radiation monitor represents the repeatability of a determination. It signifies the closeness of agreement among a number of consecutive measurements of the output for the same value of input under identical operating conditions. The reference accuracy which defines the limit that errors will not exceed when the system is used includes the combined conformity, hysteresis and repeatability errors discussed previously.

12.3.4.4 Compliance With Regulatory Guides and Standards

When applicable, the guidance provided by Regulatory Guides 8.2, 8.8, 1.21, and ANSI N13.1-1969 has been followed in the design of the area radiation and airborne radioactivity monitoring system.

1. For Regulatory Guide 8.2, the system was designed to provide information to plant operators pertaining to radiological conditions in selected plant areas and in releases of radioactive materials in effluents. Additionally, the system is designed to allow for hard copy printouts of the radiation levels measured by area monitors and radioactivity concentration levels measured by airborne monitors.

2. For Regulatory Guide 8.8, applicable guidance was followed by:

    a. Review of system design using the points outlined in the guide

    b. Proper design of this system to ensure that radiation exposures are minimized

3. Based on Regulatory Guide 1.21, the system was designed to measure and report releases of radioactive materials in gaseous effluents and to classify and report the categories and curie content of wastes.

    a. The airborne monitors detect radioactivity and give indication of effluent stream activity concentrations (detector is designed to monitor the most representative isotope of an effluent stream)

    b. Sampling programs described in Section 11.5 will be used in conjunction with the continuous monitoring system to determine total quantity of radioisotopes and types of isotopes released. Monitors will provide concentrations of sample streams by processing count rates from detectors and, where applicable, flow determinations from flow measuring devices. This information along with multichannel analysis from the sampling program will determine isotopic content. The meteorological program provides data for dose evaluation. The data and information obtained from the various subsystems and programs are used to generate the required reports according to Regulatory Guide 1.21 format.

4. The aerodynamic sampling guidelines for off-line vent stack airborne monitors provided in ANSI N13.1 are considered. Representative samples of the effluent streams are obtained by isokinetic nozzles, sample lines with no obstructions and large radii bends and appropriate sample valves.

5. Based on GDC-54, the system was designed to monitor areas which contain containment penetrations and associated piping including closed systems outside containment. See Section 6.2.5 for details.

6. Compliance with RG1.97 is described in Section 7.5.

REFERENCES:

1. Loose R. A. and Battaglia J. A., Design Considerations in PWR Waste Processing Systems, AIChE 69th National Meeting, May 16, 1971, Cincinnati, Ohio.

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3. Clemons, L., Jr., and Battaglia, J. A., Minimizing Radiation Exposures in PWR Plants, ANS Twenty-First Annual Meeting, June 8, 1975, New Orleans, La.

4. Solomon Y., and Roesmer, J., Measurement of Fuel Element Crud Deposits in Pressurized Water Reactors, Nuclear Technology, May 1976.

5. Bergman C. A., and Bohr, D. P., Radiation Level Buildup on Plant Piping During Second Core Operation of Shippingport Plant, Nuclear Technology, Vol. 13, March 1972.

6. Fletcher, W. D. Primary Coolant Chemistry of PWRs, Proceedings of the 31st International Water Conference, Pittsburgh, Pa. October 1970.

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8. Decontamination of Nuclear Plants Seminar, Ohio State University, May 1975.

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10. Engineering Compendium on Radiation Shielding, Vols. 1 and 3. Springer Verlag, Inc., New York, 1968. Sponsored by the International Atomic Energy Agency - Vienna.

11. Nuclear Data Sheets by the National Academy of Sciences. National Research Council (continuously updated).

12. “An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternative” National Environmental Studies Project (AIF/NESP-009) Industrial Forum, Inc., November 1976.

13. Design, Inspection, Operation and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Radiation Exposures As Low As Reasonably Achievable, WCAP-8872, Westinghouse Electric Corporation, April 1977.

TABLE 12.3-1
RANGE OF DF[a] VALUES MEASURED ACROSS PWR PURIFICATION DEMINERALIZER

Isotope	Demineralizer	DF Flange	
		Plant A	Plant B
Mn-54	Mixed bed[b]	5-7	-
Mn-54	Mixed bed + cation bed[c]	150-600	67-1100
Co-58	Mixed bed	140-215	-
Co-58	Mixed bed + cation bed	190-970	100-650
Co-60	Mixed bed	18-20	-
Co-60	Mixed bed + cation bed	170-1000	-
I-131	Mixed bed	300-500	-
I-131	Mixed bed + cation bed	160-380	3.5-3800
Cs-134	Mixed bed	2	-
Cs-134	Mixed bed + cation bed	195-580	470-3000
Cs-137	Mixed bed	1-2	-
Cs-137	Mixed bed + cation bed	180-1000	360-7900

a) Ratio of inlet concentration to outlet concentration

b) Generic name for demineralizer used in W system, normally in the lithium borate form; it essentially behaves as an anion bed for isotopes listed.

c) Normally in the hydrogen form

d) Taken from Reference 3

TABLE 12.3-2
APPROXIMATE RCS WETTED SURFACE AREAS AND WEIGHTS TYPICAL FOUR-LOOP PLANT

	Material	Surface Area (ft^2)	Weight (lb)
Reactor internals	SS	4236	402,000[(a)]
Reactor vessel clad	SS	2190	19,200
RCS piping	SS	2758	299,890
Reactor internal bolting materials	SS	Negligible	Negligible
RCS pumps	SS	Negligible	684,800
Auxiliary heat exchanger surfaces	Inconel	Negligible	Negligible
Steam generators	Inconel	1.90×10^5	96,800
Fuel	SS(0.12)	2000	-
Fuel	SS(0.08)	3600	-
Fuel	Inconel	7.80×10^3	-
Fuel	Zircaloy	7.78×10^4	-
Reactor internals	Stellite	3.23	-
RCS pump journals	Stellite	17.21	-
Control rod drive mechanisms	Stellite	10.76	-
RCS valves	Stellite	2.6	-

SS = Stainless Steel

a) Includes 228,300 lb of large forgings and 173,300 lb of steel plate and small forgings

TABLE 12.3-3
PRESENT SPECIFICATIONS ON COBALT CONTENT OF MATERIALS

	Material	Cobalt (w/o)
Reactor internals	SS	0.12 maximum
Reactor vessel clad	SS	0.2 maximum
RCS piping	SS	0.2 maximum
Reactor internal bolting materials	SS	0.25 maximum
RCS pumps	SS	0.2 maximum
Auxiliary heat exchanger surfaces exposed to RCS	SS	0.2 maximum
Steam generators	Inconel	0.1 maximum
Fuel	SS[(a)]	0.12 maximum
Fuel	SS[(b)]	0.08 maximum
Fuel	Inconel	0.1 maximum
Fuel	Zircaloy	0.002 maximum

SS = Stainless Steel

a) Refers to stainless steel outside active region on zircaloy clad fuel (e.g. top and bottom nozzles)

b) Refers to stainless steel inside active region.

TABLE 12.3-4
COST TO PROVIDE REDUCED COBALT CONTENT DOLLARS/TYPICAL FOUR-LOOP PLANT

	Cobalt Content (w/o)		
	0.25 percent	0.1 percent	0.05 percent
Reactor internals (in core periphery only)	$48,300	$155,000	Unavailable
Reactor vessel clad	300	2000	$4000
RCS piping	5000	31,500	66,000
Reactor internal bolting materials	Negligible	Negligible	Negligible
RCS pumps	10,300	71,900	150,600
Auxiliary heat exchanger surfaces	Negligible	Negligible	Negligible
Steam generator (four) Inconel	5800	40,700	Unavailable

Approximate Adders to Cost for Reduced Cobalt Content in Stainless Steel Plate and Small Forgings (e.g. lower core support plate)

Percent	Cents/lb
>0.25	No charge
0.25-0.15	1.5
0.15-0.10	5.25
0.10-0.05	10.5
<0.05	22

Approximate Adders to Cost for Reduced Cobalt Content in Large Stainless Steel Forgings (e.g. lower-core support plate)

Percent	Cents/lb
>0.25	No charge
0.25-0.10	20
0.10-0.05	60
<0.05	Not available

TABLE 12.3-5[a]
APPROXIMATIONS TO COBALT SOURCES AND CONTRIBUTION TO TOTAL COBALT IN RCS

Component	Surface Area (ft^2) x Cobalt Limit (w/o)	Surface Area (ft^2) x Cobalt Limit (w/o) x Corrosion Rate (mg/dm^2-mo)
Reactor internals (SS)	508	254
Reactor vessel clad (SS)	438	219
RCS piping (SS)	552	276
Steam generator tubes (Inconel)	19,000	2
Fuel (SS)	528	264
Fuel (Inconel)	780	1170
Reactor Internals (Stellite)	1.6	Not available
RCS pump journals (Stellite)	8.6	Not available
CRDM (Stellite)	5.4	Not available
RCS Valves (Stellite)	1.3	Not available
Contribution to Total (Percent)		
Stainless steel	9.28	3.30
Inconel	90.64	96.70
Stellite	0.077	
Total	100.0	100.0

SS = Stainless Steel

a) Source: WCAP-8872, Table 6-4

TABLE 12.3-6
RADIATION ZONES

Zone No.	Maximum Design Dose Rate (mrem/hr)	Design Criteria Description and Explantion
I	Less than 0.25	Uncontrolled, unrestricted access area. Nonradiation personnel and visitor occupancy: 40 hr/wk, 50 wk/yr
II	Less than 2.5	Controlled, occupational access area. Radiation worker and authorized visitor occupancy: maximum 40 hr/wk, 50 wk/yr
III	Less than 25.0	Controlled, limited access area.
IV	More than 25.0	Controlled, restricted access area with limited occupancy. Normally equipped with lockable door/gate.
V	No limitation	Controlled, restricted access area, normally inaccessible. Access is via lockable door.

TABLE 12.3-7
THIS TABLE HAS BEEN DELETED

TABLE 12.3-8
AREA RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 1 of 5)

Channel Nos. Unit 1	Unit 2	Detector Type	Monitor Location	Specified Instrument Range (mR/hr)	Bases for Alarm Set Points
Containment Building					
1RE 6290A	2RE 6290A	Ionization Chamber	Elevation 905 ft. 9 in	10^3 - 10^{10}	Note 5
1RE 6290B	2RE 6290B	Ionization Chamber	Elevation 905 ft. 9 in	10^3 - 10^{10}	Note 5
1RE 6251	2RE 6251	G-M Tube	Elevation 860 ft. 0 in	10^{-1} - 10^4	Note 6
1RE 6253	2RE 6253	G-M Tube	Elevation 860 ft. 0 in.	10^{-1} - 10^4	Note 6
1RE 6255	2RE 6255	Ionization Chamber	Elevation 808 ft. 0 in.	10^2 - 10^7	Note 6
1RE 6256	2RE 6256	Ionization Chamber	In-core instrumentation room Elevation 849 ft. 0 in.	10^2 - 10^7	Note 6
1RE 6285	2RE 6285	Ionization Chamber	Seal table room, Elevation 831 ft. 6 in.	10^2 - 10^7	Note 6

TABLE 12.3-8
AREA RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 2 of 5)

Channel Nos. Unit 1	Unit 2	Detector Type	Monitor Location	Specified Instrument Range (mR/hr)	Bases for Alarm Set Points
Safeguards Building					
1RE 6259	2RE 6259	G-M Tube	Filter Storage Area Elevation 873 ft. 6 in.	10^{-1} - 10^{4}	Note 2
1RE 6259A	2RE 6259A	Ionization Chamber	Mechanical Penetration Area Elevation 810 ft. 6 in.	10^{2} - 10^{7}	Note 3
1RE 6259B	2RE 6259B	Ionization Chamber	Mechanical Penetration Area Elevation 810 ft. 6 in.	10^{2} - 10^{7}	Note 3
1RE 6260A	2RE 6260A	Ionization Chamber	RHR Pump Room Elevation 773 ft. 0 in.	10^{2} - 10^{7}	Note 4
1RE 6260B	2RE 6260B	Ionization Chamber	RHR Pump Room Elevation 773 ft. 0 in.	10^{2} - 10^{7}	Note 4
1RE 6261	2RE 6261	GM Tube	Sampling Room Elevation 810 ft. 6 in.	10^{-1} - 10^{4}	Note 2
1RE 6291A	2RE 6291A	Ionization Chamber	Valve isolation tank room Elevation 790 ft. 6 in.	10^{2} - 10^{7}	Note 3

TABLE 12.3-8
AREA RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 3 of 5)

Channel Nos. Unit 1	Unit 2	Detector Type	Monitor Location	Specified Instrument Range (mR/hr)	Bases for Alarm Set Points
1RE 6291B	2RE 6291B	Ionization Chamber	Valve isolation tank room Elevation 790 ft. 6 in	10^2 - 10^7	Note 3
1RE 6292	2RE 6292	Ionization Chamber	Switchgear Room Elevation 810 ft. 6 in.	10^2 - 10^7	Note 3
1RE 6293	2RE 6293	Ionization Chamber	Piping Penetration Area Elevation 831 ft. 6 in.	10^2 - 10^7	Note 3
1RE 6294	2RE 6294	Ionization Chamber	Electrical Equipment Room Elevation 831 ft. 6 in.	10^2 - 10^7	Note 3
1RE 6295	2RE 6295	Ionization Chamber	Containment Access Hall Elevation 831 ft. 6 in.	10^2 - 10^7	Note 3
1RE 6296	2RE 6296	Ionization Chamber	Switchgear Room Elevation 852 ft. 6 in.	10^2 - 10^7	Note 3
1RE 6297	2RE 6297	Ionization Chamber	Emergency airlock Elevation 896 ft. 6 in.	10^2 - 10^7	Note 3

TABLE 12.3-8
AREA RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 4 of 5)

Channel Nos. Unit 1	Unit 2	Detector Type	Monitor Location	Specified Instrument Range (mR/hr)	Bases for Alarm Set Points
Auxiliary Building					
Fuel Building					
XRE 6272	XRE 6273	G-M Tube	Spent Fuel Pool Operating floor Elevation 860 ft. 0 in.	10^{-1} - 10^{4}	Note 1
XRE 6274	XRE 6275	G-M Tube	Spent Fuel Pool Operating floor Elevation 860 ft. 0 in.	10^{-1} - 10^{4}	Note 1
Turbine Building					
XRE 6283		G-M Tube	Hot lab area Elevation 810 ft. 6 in.	10^{-1} - 10^{4}	Note 2

TABLE 12.3-8
AREA RADIATION MONITORING SYSTEM PARAMETERS

(Sheet 5 of 5)

Channel Nos. Unit 1	Unit 2	Detector Type	Monitor Location	Specified Instrument Range (mR/hr)	Bases for Alarm Set Points
Electrical and Control Building					
XRE 6281	XRE 6282	G-M Tube	Control Room Elevation 830 ft. 0 in.	10^{-1} - 10^{4}	Note 2

Notes:

1. Alarm set points are based on personnel occupancy for normal operation.
2. Alarm set points are based on personnel occupancy for accident conditions.
3. Alarm set points are based on verification of containment integrity for accident conditions.
4. Alarm set points are based on long term equipment surveillance for accident conditions.
5. Alarm set points are based on requirements for emergency planning.
6. Alarm set points are based on surveillance of restricted areas for normal operation.

TABLE 12.3-9
AREA RADIATION MONITORING SYSTEM ENVIRONMENTAL DESIGN CONDITIONS

Item			Normal	Accident	Normal	Accident	Normal	Accident
a.	Control room electronics		80	80	50	50	0.1	0.1
b.	Detectors and Local electronics							
	1)	Outside containment	122	122	95	100	-0.1	ATM
	2)	Inside containment (non-1E)	122	300 (3 hours)	70	Steam, air, and water mixture	ATM	50 psig
	3)	Inside containment (1E)	122	333	70	100	ATM	50 psig

Note:

ATM - atmospheric

12.4 DOSE ASSESSMENT

The following discussion includes historical estimates of personnel occupancy in plant radiation zones and personnel radiation exposures expected annually. The discussion includes a basis for, and an estimate of, the total annual in-plant personnel radiation exposure expected for CPNPP operations. These estimates were prepared as part of the CPNPP Operating License application prior to Unit 1 operation and will not be further maintained in this FSAR Section. Luminant Power maintains records of actual in-plant personnel radiation exposures and routinely provides various reports of personnel exposure data, including annual cummulative person-rem totals, as part of the ongoing operational Radiation Protection Program.

12.4.1 ESTIMATED PLANT OCCUPANCY REQUIREMENTS

Table 12.4-1 presents estimated station personnel occupancy requirements of plant radiation areas (radiation zones II, III, IV, and V) during normal operation and anticipated operational occurrences.

12.4.2 DESIGN DOSE RATES

The zone descriptions presented in Section 12.3.1.3 and Table 12.3-6 outline the design maximum dose rate limits for the design of shielding in zones I, II, and III during normal operation, shutdown, and refueling. These limits are 0.25, 2.5, and 25 mrem/hr, respectively. For in-plant areas, these maxima are not expected to occur because plant shielding and access control are based on radionuclide activities corresponding to one percent failed fuel. Because fuel failure and associated radionuclide concentrations are considerably less than those used on a design basis, the maximum expected dose rates in these areas would, likewise, be considerably less. All in-plant areas are designed so that occupational radiation exposure (ORE) is ALARA. During maintenance and inspection in zone IV areas, ORE is kept ALARA by using administrative procedures, drainage, purging, flushing, backwashing, and decontamination. Occupancy in high-radiation areas is limited by administrative control procedures based on routine radiation survey measurements.

12.4.3 ESTIMATED MAN-REM EXPOSURES FOR MAJOR PLANT FUNCTIONS

Total in-plant exposures from 26 operating pressurized water reactor (PWR) units were reported in Reference [1], for 1975, as an average of 309 man-rems with a standard deviation of 275 man-rems. Using the reported average value of 309 man-rems, and normalizing to CPNPP capacity per reactor (1150 MWe, nominal), yields an estimated total in-plant exposure of 421 man-rems per year.

In 1975, an average of 578 persons per reactor received measurable exposures, as reported in Reference [1]. Based on the above value for total dose (421 man-rem), total number of in-plant personnel exposed (578), and distribution data from Table 5 of Reference [1], the distribution of personnel and man-rem, and number of personnel and cummulative man-rem, may be estimated for various in-plant work functions. This data is provided in Tables 12.4-2 and 12.4-3, respectively. Estimates of man-rem doses during refueling are presented in Table 12.4-4. The estimates provided are the expected exposures to an experienced refueling crew; however, refueling operations where trouble is encountered can result in much higher exposures.

In 1975, the average exposure per individual was reported as 0.8 rem [1]. The average exposure per individual for maintenance and refueling activities was 0.77 rem, whereas the average exposure per individual for reactor operations was 0.91 rem. This latter value is a factor of 5.5 below the 5-rem/yr limit presented in 10 CFR Part 20 upon which the 100-mrem/week design basis is established. Therefore, the design approach used on similar plants in which the 100-mrem/week value is used would result in operational levels of individual exposure, a factor of 5.5 below the 10 CFR Part 20[a] limits.

Because most of the anticipated exposures occur under circumstances which are difficult to predict analytically, such as maintenance on radioactive components, shielding design is based on worst case assumptions, and design features are provided which reduce exposures. So, actual exposures should be consistent with the ALARA approach.

The estimated total in-plant exposure of 421 man-rems shown in Table 12.4-3 is within the range of values reported in Reference [1], in 1975, and is not expected to be exceeded by long-term exposures. This value is representative of the average annual exposure during the station lifetime.

12.4.4 ESTIMATED DOSE TO CONSTRUCTION WORKERS

The total dose to construction workers at the site during power operation is the sum of dose contributions from gaseous releases and direct and scattered radiation from contained sources.

12.4.4.1 Construction Worker Dose as a Result of Gaseous Releases

In addition to radiation fields in the general construction area as a result of direct radiation from all Unit 1 sources, there is a contribution to construction worker dose from airborne radionuclides released from the normal operation of Unit 1. These doses, resulting from releases, are calculated assuming the release rates shown in Table 11A-4 of the FSAR. The X/Q values used in estimating exposures in the Unit 2 construction area have been estimated using the straight-line diffusion of NRC Regulatory Guide 1.111, as applied in Appendix 11A of the FSAR, and are presented in Table 12.4-5. All transport times, dose conversion factors and other constants used were those suggested in NRC Regulatory Guide 1.109. Distances and directions from the Unit 1 Containment are plotted as shown on Figure 12.4-1, Building Complex Layout. The corners of the Unit 2 building under construction are chosen arbitrarily because these extremities offer the most vulnerable locations for construction personnel to work.

The location and number of workers during the estimated 16-month Unit 2 construction period are listed in Table 12.4-6. Individual predicted doses are presented at 10 potential worker sites of interest and at 16 site boundary points at the midpoint of each affected wind sector. It is assumed that construction workers spend 2500 hr/yr at the jobsite (50 hr/week for 50 weeks/yr). Also factored into the resultant expected doses shown in Table 12.4-7 is the fact that the breathing rate for occupational exposures is taken to be 10 m^3 per 8-hr period vs. 20 m^3 adult inhalation rate for non-occupational individuals for a 24-hr period. Results are tabulated for total body and skin doses for cloud submersion and total body and thyroid inhalation doses. Estimates are presented on an annual basis (1981 period) such that the 4-man-rem dose

a. Design assessment made with the provisions of 10 CFR 20.1-20.601.

estimates for cloud submersion are derived by exposing the average numbers of workers at each building under construction, shown in Table 12.4-6, to the worst computed individual dose for that particular structure. The results presented in Table 12.4-8 distinguish the 4-month period in 1980 from the 1981 1-yr period.

The estimated doses resulting from airborne radioactivity received by construction workers on Unit 2 as a result of the operation of Unit 1 are well within the limits of 10 CFR Part 20 for exposure of occupationally employed individuals in unrestricted areas.

12.4.4.2 Estimated Dose To Construction Workers Resulting From Contained Sources

The greater part of the construction activity that occurs after the startup of Unit 1 is performed inside buildings. For the purpose of the following calculations, it is assumed that workers spend 15 percent of their workday outdoors. A dose reduction factor of approximately 5.5 (see Section 12.4.3) is applied to the maximum design dose rates of radiation zones (see Section 12.3.1.3) corresponding to the construction activities indoors near contained sources (taken to be Zone II, i.e., 2.5 mrem/hr) and construction activities outdoors and indoors not near contained sources (taken to be Zone I, i.e., 0.25 mrem/hr). The resulting estimated average annual exposures (based on 50 weeks/year and 50 hours/week occupancy) are 1.14 rem/year and 0.11 rem/year, respectively.

Estimates of construction manpower requirements following startup of Unit 1, based on data presented in Table 12.4-6, are shown in Table 12.4-9. Anticipated annual exposures at various construction locations are presented in Table 12.4-10. Total construction worker ORE expressed in man-rems is the product of manpower (in man-yr) and the anticipated annual exposure (in rem/yr). Results of this calculation are presented in Table 12.4-11. These results may be summarized as follows:

1. Over the period considered, the total exposure to the total construction force resulting from the contained sources is 297.71 man-rems.

2. The average annual exposure to an individual is 0.46 rem/yr (297.71 man-rem/ 653.33 man-yr).

3. The average annual exposure to an individual working in the Auxiliary Building, Fuel Building, or Safeguards Building, Unit 2 (after fuel load of Unit 2) is 0.98 rem/yr (0.85 x 1.14 rem/yr + 0.15 x 0.11 rem/yr).

4. The highest exposure received by a construction worker during the entire period considered is 1.31 rems (0.98 rem/yr x 1.33 yr).

12.4.5 ESTIMATED ANNUAL DOSE AT THE EXCLUSION BOUNDARY AND TO THE POPULATION AT LARGE

At the minimum exclusion boundary distance the estimated annual exposure to an individual as a result of contained sources is approximately 3.25×10^{-3} mrems. The estimated additional annual exposure due to the ISFSI with 84 casks is well below 1 mrem. The total exposure to the cumulative population within 50 miles of the site (for census year 2000) as a result of contained sources is approximately 1.53×10^{-3} man-rem.

REFERENCES

1. Murphy, T.D., et al., Occupational Radiation Exposure at Light Water Cooled Power Reactors, 1969-1975, U.S. Nuclear Regulatory Commission, NUREG-0109, August 1976.

TABLE 12.4-1
ESTIMATED STATION PERSONNEL OCCUPANCY OF PLANT RADIATION AREAS (MAN-HOURS/YEAR)

	Radiation Zone					
Personnel	I	II	III	IV	V	Totals
Operators	40,000	25,000	3,600	950	250	120,000
Maintenance	52,000	42,000	4,800	1,080	120	100,000
Technical Support	80,000	35,000	4,000	900	100	120,000
Administrative	54,000	6,000	(a)	(a)	(a)	60,000

a) Denotes insignificant occupancy

TABLE 12.4-2
DISTRIBUTION OF PERSONNEL AND COLLECTIVE DOSE EQUIVALENTS
FOR VARIOUS WORK FUNCTIONS

Work Function	Percentage of Total Personnel	Percentage of Total Man-Rem
Reactor Operations And Surveillance	9.2	10.8
Routine Maintenance	59.5	52.6
Inservice Inspection	4.1	3.0
Special Maintenance	16.3	19.0
Waste Processing	7.6	6.9
Refueling	3.3	7.7

TABLE 12.4-3
ESTIMATES OF PERSONNEL AND COLLECTIVE DOSE EQUIVALENTS FOR VARIOUS WORK FUNCTIONS

(Sheet 1 of 3)

Work Function			Est. Dose Range mrem/hr.	Duration man hrs/yr	Dose Man-Rem
1)	Reactor Operations & Surveillance (estimated number of personnel = 53)				
	a)	Routine Patrols and Operations	.25/300	18,000	8
	b)	Radiation Surveys	.25/2.5	5,500	6
	c)	Radiation Surveys	2.5/100	4,000	13
	d)	Radiation Surveys	>100	50	7
	e)	Periodic Tests, Insp. & Calib.	.25/2.5	10,000	3
	f)	Periodic Tests, Insp. & Calib.	2.5/25	1,500	4
	g)[a]	Control Room Operations	<.25	50,000	5
					46
2)	Routine Maintenance (estimated number of personnel = 344)				
	a)	Normal Operations			
		1) Pumps	.25/50	2,500	5
		2) Valves	.25/50	4,000	10
		3) Instrument & Controls	.25/25	4,000	3
		4) Motors	.25/25	1,000	2
		5) Heat Exchangers	.25/1500	1,000	4
		6) Waste Processing Systems	.25/1500	1,000	7
		7) Demineralizer (Resin Change)	2.5/25	1,000	3
		8) Filter Changes	2.5/25	1,000	3
		9) HVAC	.25/25	500	1
		10) Misc.	.25/25	1,500	6

TABLE 12.4-3
ESTIMATES OF PERSONNEL AND COLLECTIVE DOSE EQUIVALENTS FOR VARIOUS WORK FUNCTIONS

(Sheet 2 of 3)

Work Function				Est. Dose Range mrem/hr.	Duration man hrs/yr	Dose Man-Rem
	b)	Refueling Operations				
		1)[b]	Preparation for Head Removal	5/200	1,900	40
		2)	Head Removal	5/200	80	11
		3)	Upper Internals Removal	2.5/150	100	2
		4)	Installation of Upper Internals	2.5/150	100	2
		5)[b]	Preparation for Head Installation	5/1500	1,900	45
		6)	Head Installation	5/200	80	11
		7)	Refueling Cavity Clean-Up	5/100	500	10
		8)	Preparation for Containment Closure	2.5/100	2,800	36
		9)	Steam Generator Test	5/15,000	1,000	10
		10)	Incore Instrumentation	10/50	500	10
						221
3)	Inservice Inspection (estimated number of personnel = 24)			2.5/200	700	13
4)	Special Maintenance (estimated number of personnel = 94)					
	a)	Steam Generator Maintenance		5/15,000	4,000	22
	b)	Reactor Coolant Pump Seal Inspection and Repair		5/500	1,000	8
	c)	Evaporator		5/100	500	5
	d)	Heat Exchanger		5/100	700	10
	e)	Demineralizer		5/100	500	6

TABLE 12.4-3
ESTIMATES OF PERSONNEL AND COLLECTIVE DOSE EQUIVALENTS FOR VARIOUS WORK FUNCTIONS

(Sheet 3 of 3)

Work Function			Est. Dose Range mrem/hr.	Duration man hrs/yr	Dose Man-Rem
	f)	Safety System Modification	5/100	500	5
	g)	Pumps	.25/100	700	6
	h)	Waste Processing System	.25/1500	500	7
	i)	Motors	.25/25	500	5
	j)	Instrument & Control	.25/25	300	3
	k)	HVAC	.25/25	500	3
					80
5)	Waste Processing (estimated number of personnel = 44)		2.5/100	4,400	29
6)	Refueling (estimated number of personnel = 19)				
	a)	Health Physics Operations	.25/1500	7,000	14
	b)	Fuel Handling	.25/25	1,300	12
	c)	Testing & Inspections Specific to Refueling	2.5/100	6,000	6
					32
TOTALS		(estimated number of personnel = 578)			421

a) The control room is a Zone I radiation area, but it is included in this table due to the large duration associated with it.

b) Items (1) and (5) include detensioning/removal of reactor stud bolts and installation/ tensioning of same

TABLE 12.4-4
ESTIMATES OF COLLECTIVE DOSE EQUIVALENTS DURING REFUELING

Operation	Average Radiation Field (mrem/hr)	Man-Rem
Missile shield	25	0.40
Reactor vessel head connections	60	4.60
Cavity seal ring	50	0.00
Reactor vessel stud work	80	10.24
Guide studs and stud hole plugs	80	1.28
Reactor vessel head lift and cavity fill/drain	15	1.05
CRDM drive shafts	15	0.72
Upper internals lift	15	0.72
Fuel shuffle	15	12.00
TOTAL		31.73

TABLE 12.4-5
EXPECTED ATMOSPHERIC RELATIVE CONCENTRATION VALUES AT CALCULATION LOCATIONS

Building/ Location	Distance (m)	Sector	X/Q (sec m^{-3})
Auxiliary Building	105	NW	9.00E-04
" "	81	W	5.60E-04
" "	50	WNW	2.28E-03
" "	66	N	1.92E-03
Containment, Unit 2	89	N	1.08E-03
Safeguards Building, Unit 2	87	NW	1.28E-03
" "	131	NNW	6.40E-04
" "	122	N	5.80E-04
Fuel Building	71	NNE	1.04E-03
" "	37	ENE	2.24E-03

TABLE 12.4-6
LOCATION AND NUMBER OF CONSTRUCTION WORKERS[a]

Building	1980 Average for 4 Months	1981 Average for 12 Months
Containment, Unit 2	249	179
Auxiliary Building	99	96
Fuel Building	25	25
Safeguards Building, Unit 2	201	162
Average for period	574	462

a) Includes preoperational testing and start-up personnel.

TABLE 12.4-7
INDIVIDUAL EXPECTED CONSTRUCTION WORKER DOSE EQUIVALENT[a]

Building/ Location	Distance (m)	Affected Sector	Total Body[b] Cloud Dose (mrem/yr)	Cloud[b] Skin Dose (mrem/yr)	Total Body[b][c] Inhalation Dose (mrem/yr)	Thyroid[b][c] Inhalation Dose (mrem/yr)
Auxiliary Building	105	NW	4.65E+00	1.32E+01	1.68E+01	2.08E+01
" "	81	W	2.09E+00	8.20E+01	1.04E+01	1.30E+01
	50	WNW	1.18E+01	3.34E+01	4.24E+01	5.25E+02
	66	N	9.80E+01	2.78E+01	3.53E+01	4.41E+01
Containment, Unit 2	89	N	5.60E+01	1.58E+01	2.01E+01	2.50E+01
Safeguards Bldg. Unit 2	87	NW	6.60E+01	1.87E+01	2.38E+01	2.98E+01
" "	131	NNW	3.31E+00	9.35E+00	1.19E+01	1.48E+01
	122	N	3.00E+00	8.50E+01	1.08E+01	1.34E+01
Fuel Building	71	NNE	5.40E+00	1.53E+01	1.94E+01	2.42E+01
" "	37	ENE	1.16E+01	3.28E+01	4.18E+01	5.20E+02

a) In the tables "Dose" is used for "Dose Equivalent"

b) Workers spend 50 hr/week, 50 weeks/yr on the job.

c) Occupational inhalation rate 10 m^3/8-hr day or 30 m^3/24-hr period vs. 20 m^3 adult breathing rate assumed for nonoccupation exposure.

TABLE 12.4-8
ESTIMATION OF COLLECTIVE DOSE EQUIVALENT TO CONSTRUCTION WORKERS DURING 1980 TO 1981 PERIOD

	No. of Workers (1980)		No. of Workers (1981)	
Building	Average for 4 Months	Man-Rems	Average for 12 Months	Man-Rems
Containment, Unit 2	249	4.63E-01	179	1.00E+00
Auxiliary Building	99	3.89E-01	96	1.13E+00
Fuel Building	25	9.65E-01	25	2.90E-01
Safeguard Building, Unit 2	201	4.42E-01	162	1.07E+00
Average for period	574	1.39E+00	462	3.49E+00

TABLE 12.4-9
CONSTRUCTION MANPOWER REQUIREMENTS

Location	Manpower (man-yr)	
	Inside	Outside
Containment, Unit 2	222.70	39.30
Auxiliary Building	109.65	19.35
Fuel Building	28.33	5.00
Safeguards Building Unit 2 (prior to fuel load of Unit 2)	114.33	20.18
Safeguards Building Unit 2 (after fuel load of Unit 2)	80.33	14.18
Subtotals	555.33	98.00
TOTAL		653.33

TABLE 12.4-10
ANNUAL DOSE EQUIVALENT TO CONSTRUCTION WORKERS RESULTING FROM CONTAINED SOURCES

Location	Annual Dose Equivalent (rem/yr)	
	Inside	Outside
Containment, Unit 2[(a)]	0.11	0.11
Auxiliary Building	1.14	0.11
Fuel Building	1.14	0.11
Safeguards Building Unit 2 (prior to fuel load of Unit 2)	0.11	0.11
Safeguards Building Unit 2 (after fuel load of Unit 2)	1.14	0.11

a) All work assumed to be completed prior to fuel load of Unit 2.

TABLE 12.4-11
CONSTRUCTION WORKER OCCUPATIONAL COLLECTIVE DOSE EQUIVALENT
RESULTING FROM CONTAINED SOURCES

	ORE (man-rem)	
Location	Inside	Outside
Containment, Unit 2	25.32	4.47
Auxiliary Building	124.69	2.20
Fuel Building	32.21	0.57
Safeguards Building, Unit 2 (prior to fuel load of Unit 2)	13.00	2.29
Safeguards Building, Unit 2 (after fuel load of Unit 2)	91.34	1.61
Subtotals	286.57	11.14
TOTAL		297.71

12.5 RADIATION PROTECTION

12.5.1 ORGANIZATION

The CPNPP radiation protection program is established to provide an effective means of radiation protection for station personnel, visitors, and the general public. To provide this, the radiation protection program incorporates a dedicated philosophy from management (Section 12.1.1), qualified personnel to direct and to implement the radiation protection program, sufficient equipment, facilities, written procedures and instructions based upon acceptable radiation protection practices and guidance. The CPNPP operational radiation protection program is developed and implemented through the applicable guidance of Regulatory Guides; 8.2 (February 1973); 8.8, Revision 3 (June 1978); and 8.10, Revision 1-R (May 1977).

The Radiation Protection Department implements and enforces the CPNPP radiation protection program. The Manager, Nuclear Radiation Protection, who reports directly to the Plant Manager (see Section 13.1.1.2.1), is responsible for oversight and implementation of the radiation protection program. The Manager, Nuclear Radiation Protection, directs and supervises radiation protection personnel and activities in the plant. The ultimate responsibility of the radiation protection program lies with the Plant Manager.

The Manager, Nuclear Radiation Protection is responsible for insuring compliance with regulatory requirements in the following areas:

1. Shipping and receiving of all radioactive material.

2. Transfer and storage of radioactive waste.

3. Approval of all radioactive waste containers.

4. Collection, sorting, and packaging of radioactive waste.

Qualified Radiation Protection technicians are assigned to the station and are familiar with station lay-out, personnel, procedures, and equipment. Additional information on the qualifications and experience of the Manager, Nuclear Radiation Protection and technicians can be found in Section 13.1 of the FSAR. Radiation Protection supports the Operations, Maintenance, and Engineering Departments and provides radiation protection coverage for activities that involve exposure to radiation or radioactive material. During major maintenance and refueling outages, additional qualified technicians may be secured, as required, based on the work load. In addition, Radiation Protection is organized to provide the following services:

1. Developing radiation protection procedures and instructions for routine and non-routine activities that may be encountered in operating, maintaining, inspecting, and testing the station.

2. Ensuring that the provisions and standards of 10 CFR Part 20 are implemented.

3. Providing a personnel radiation dosimetry program and maintaining dosimetry records.

4. Performing radiological surveys of station areas and maintaining records of survey results.

5. Assisting in the station training program by assuring that adequate radiation protection training is provided.

6. Maintaining an adequate supply of calibrated radiation detection instruments and equipment for assessing the radiological conditions at CPNPP.

7. Providing, maintaining, and issuing protective clothing and equipment.

8. Preparing and issuing reports dealing with the radiological aspects of CPNPP, as required by regulatory agencies and station procedures.

9. Directing the decontamination of personnel, equipment, and facilities at CPNPP.

10. Developing, implementing and maintaining the CPNPP radiological respiratory protection program.

It is the responsibility of each individual to obey radiological control procedures and to report to Radiation Protection personnel any circumstances where procedures are not being followed or unsafe activities are occurring. Each individual assigned to work in Radiologically Controlled Areas (RCA) is responsible for demonstrating a thorough working knowledge of radiation protection rules and procedures. Personnel requiring access to radiation areas are trained in radiation protection principles and are responsible for demonstrating proficiency in radiation protection skills. New employees required to work in RCAs are required to complete the training program and demonstrate their proficiency prior to beginning work assignments. Visiting or contract individuals will be escorted in the RCA until they demonstrate an adequate knowledge of radiation protection principles through the completion of the training program.

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

Radiation Protection procures and maintains the appropriate monitoring instrumentation and protective equipment and supplies necessary for operating and maintaining the CPNPP radiation protection program. Additionally, adequate facilities are available for the conduct of radiation protection activities such as controlling access to Radiologically Controlled Areas, locating and storing equipment, performing calibrations, etc.

Equipment, facilities and instruments necessary to implement the Radiation Protection Program are highly reliable, commercial grade items. The operation, maintenance and calibration of this equipment is subject to the Operations Quality Assurance Program. Activities necessary to implement the Radiation Protection Program are controlled by approved procedures and are subject to proper Quality Assurance controls and surveillances. The following sections provide information on the types of equipment available, including criteria for selection, and radiation protection facilities.

12.5.2.1 Radiation Detection Instrumentation and Calibration Facilities

Portable, non-portable and laboratory radiation detection equipment are available to provide the appropriate detection capabilities, ranges, sensitivities, and accuracies required for the anticipated types and levels of radiation to be found at CPNPP, whether during normal operations and maintenance, or emergency conditions. The minimum inventories of instruments and equipment is listed in Table 12.5-1. This table inventory includes equipment assigned for

use in both normal plant operations and emergency preparedness. Some equipment assigned for use in plant operations may periodically exceed the designated instrument calibration date depending on circumstances, e.g., period after an outage with no intention of use. However, all equipment is considered available and is calibrated prior to use in accordance with calibration schedules and station procedures. Sufficient inventories of instrumentation are maintained to support radiation protection programs and activities for plant operations and maintenance. Non-portable and laboratory equipment is available to support radiation protection and radiochemistry functions, including a gamma spectroscopy system, a liquid scintillation counter, a low-background proportional counter, whole body counting systems, and personnel contamination monitors. Functional checks are performed on each counting system in accordance with procedures or instructions in order to determine such things as instrument response, background count rates and counting efficiencies. Records are maintained for each instrument or counting system. Repair and maintenance of this equipment is performed by station or contract personnel. Available portable instrumentation includes: low and high range ion chambers; G-M survey meters, smear counters and friskers; alpha scintillation survey meters; neutron dose-equivalent rate meters; continuous air monitors; and airborne grab samplers.

Storage locations for radiation protection instrumentation will depend on the function and frequency of use of the instrument. In general, such instrumentation is stored at locations that offer convenient accessibility to the work areas, such as the access control office.

Normally, calibration of portable and non-portable radiation protection equipment is performed onsite by station personnel. Calibration instructions are written with detailed records of calibration and maintenance of each instrument maintained at the station.

Calibration of portable radiation protection equipment is performed in the Radiation Protection Instrumentation Calibration Facility (or other suitable location) or by a qualified vendor. The Radiation Protection Instrumentation Calibration Facility is located in the Turbine Building Annex adjacent to the Unit 1 Turbine Building. This facility is conveniently located and provides a low background environment, a storage area for calibration sources, and adequate work space. Calibrations are performed using radioactive sources traceable to the National Institute of Standards and Technology (NIST) or using transfer instruments, such as electrometers, which have been calibrated using NIST traceable sources. Each instrument is labeled with a calibration sticker after calibration.

The Radiation Protection Instrumentation Calibration Facility includes a high activity Cs-137 source (initially about 5,000 Ci in 1983) housed in a calibration well assembly. The well assembly provides shielding for the source and consists of a 30 foot (below grade) vertical well containing the calibration source which rests on an elevator assembly.

The well assembly is designed to ensure that radiation exposures to individuals in unrestricted areas are in compliance with 10CFR Part 20. An elevator operating assembly, consisting of a chain and sprocket assembly with braking ability and a device to indicate the position of the source, is used to raise, lower or hold the vertical position of the well source elevator assembly. The calibration well assembly also includes five moveable shields mounted in tracks that are used for establishing desired exposure rates. These shields are locked to shield the active area above the calibration well during periods when the Calibration Well Source is not in use. Each shield is also equipped with a handle which allows the operator to move the shields along their

tracks without extending any portion of the body into or over the active area of the Calibration Well.

During periods when the source is in use, a high radiation area exists on the roof of the calibration facility. This area is fenced to prevent access, posted with appropriate warning signs and equipped with a warning light that is energized when the source is in use. There is no formal or routine access to the roof of the facility.

The following physical and administrative controls are used to prevent inadvertent exposures resulting from the Calibration Well Source:

1. During periods when the Calibration Facility is not attended, the source is lowered to its bottom position, and the physical barriers designed to prevent access to or operation of the Calibration Well Source are locked. These physical barriers include locking the elevator assembly hand wheel, locking the moveable shields, and locking the entrance to the well source room.

2. Warning lights are mounted on the roof of the calibration facility and in the well source room. These lights are activated by any movement of the moveable shields from their stored position or whenever the well source is raised off the bottom of the calibration well.

3. The well source assembly contains an upper limit stop which prevents the source from being raised above a designated height that would result in excessive exposures to the operator and in areas outside of the calibration facility.

4. All operations of the Calibration Well Source are performed in accordance with approved station procedures.

5. All activities involving use of the Calibration Well Source require the issuance of a Radiation Work Permit.

6. Use of the Calibration Well Source is only allowed by personnel specifically trained and qualified in its use.

12.5.2.2 Radiation Protection, Access Control, Decontamination and Laboratory Facilities

Radiation Protection and Access Control facilities are located in the Turbine Building on Elevation 810'-6". These areas provide office space for Radiation Protection personnel and are arranged to maintain positive control for entry and exit of personnel and equipment to and from the Radiologically Controlled Areas (RCA).

Access Control Facilities are used to facilitate access into and out of the RCA. Male and female dressing areas are located in the access control areas. These dressing areas are used for donning modesty clothing prior to RCA entry; therefore, separate change areas within the RCA are not necessary. Information necessary for entry into the RCA is available in the access control areas. Normal information reviewed in this area may include the RWP/GAP and the radiological conditions of the room or building in which personnel may be working. Personnel log into the RCA at the Access Control Point located on the 810'-6" elevation. The individual's GAP/RWP authorization, whole body count, current Radiation Worker Training and GAP/RWP requirements are verified prior to access.

Upon completion of RCA entry, all items and personnel must be monitored for contamination prior to exit and release from the RCA.

A Personnel Decontamination room is available near the Access Control Point to decontaminate workers. The decontamination room is supplied with protective clothing, cleaning agents and decontamination facilities (i.e. sink, showers, etc.). A Personnel Decontamination room is normally used after radioactive contamination has been detected at a monitoring location inside the primary RCA or during RCA exit at the Access Control Point. Drains from the decontamination room are routed to the Liquid Waste Processing System.

Personnel exit and log out of the RCA through the Access Control Point after monitoring themselves for contamination. Other exit processing may include collection of dosimetry and any special equipment that may have been issued. After exiting the Access Control Point, personnel may return to the appropriate locker room to change from modesty clothing back to personal clothing.

A low-background Chemistry counting room is available and is used for counting and/or identifying radioactivity in samples. Work activities in the counting room are controlled by station procedures. These procedures are based upon acceptable radiation protection techniques and philosophy. The counting room is equipped with a gamma spectroscopy system, liquid scintillation detector, proportional counter and other assorted counting and ancillary equipment. A hot laboratory is available where primary side samples are processed and prepared for counting. The sink and floor drains in this room are directed to the Liquid Waste Processing System and the fume hood exhaust is directed to the ventilation system. Equipment included in this room consists of such items as survey meters, and miscellaneous laboratory instruments such as pH meters, spectrophotometer, conductivity meter, and various glassware and chemicals.

12.5.2.3 Protective Clothing

Protective clothing is prescribed by Radiation Protection based upon the actual or potential radiological conditions expected for the job assignment. Protective clothing issue stations are established and maintained at inplant locations required to ensure efficient operations and to preclude the spreading of contamination. Used protective clothing is normally laundered by contracted vendor services.

12.5.2.4 Respiratory Protection Equipment

Respiratory protection equipment is available to station personnel and is prescribed and issued to individuals as required by the actual or potential radiological conditions of the work assignment. Selection of respiratory protection equipment such as full-face masks, self-contained breathing apparatus, and chemical cartridge respirators is made following the guidance of applicable regulations contained in 30 CFR Part 11 and the National Institute for Occupational Safety and Health's (NIOSH) Certified Equipment Manual. Respiratory devices available at CPNPP include the following:

1. Full-face masks with high-efficiency particulate and charcoal filters.

2. Full-face masks with airline respirator.

3. Hoods and suits with airline respirator.

4. Full-face masks with self-contained breathing apparatus.

Respiratory protection equipment is stored at the Access Control area, Control Room (SCBAs only), EOF (SCBAs only), and Fuel Building. The Respiratory Protection Program is described in detail in Section 12.5.3.6.

12.5.2.5 Personnel Exposure Monitoring Equipment

Thermoluminescent Dosimeter (TLD) Badges or Electronic Alarming Dosimetry (EAD), and Pocket Ion Chambers (PICs) are available in accordance with plant procedures for monitoring external radiation exposure to station personnel, visitors, and support personnel who enter Radiologically Controlled Areas at CPNPP. Neutron dosimetry devices are also available for those individuals subject to neutron exposure as required by job assignment and location. TLD multi badges and extremity dosimeters will be available for those assignments that require them. The TLD Badges are processed and a definitive dose evaluation is provided by the approved service organization as requested.

Whole body counters and personnel contamination monitors are used for in-vivo radiation measurements of station personnel, visitors, or support personnel to assess internal radiation exposure. Whole body counters and personnel contamination monitors will provide preliminary background information, periodic evaluation, and emergency capability for detecting internal exposure conditions.

The CPNPP personnel exposure monitoring program is described in detail in Section 12.5.3.5.

12.5.3 PROGRAMS AND PROCEDURES

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure. The radiation protection programs, procedures, and instructions developed for Comanche Peak Nuclear Power Plant are an integral part of the ALARA policy as discussed in Section 12.1. These procedures, utilized by well-trained and qualified personnel, should contribute significantly to the overall reduction of the occupational radiation exposures. The provisions and suggestions of Regulatory Guides 8.8, Revision 3 (June 1978), and 8.10, Revision 1-R (May 1977), are closely followed in the development of the radiation protection procedures and instructions. The facilities and equipment necessary to support these programs, procedures, and instructions are discussed in Section 12.5.2.

12.5.3.1 Radiation Surveillance

Radiation Protection routinely surveys selected areas at CPNPP for the assessment of radiation-field, radioactive contamination, and airborne radioactivity levels. Portable instruments, equipment, and techniques used for these surveys are selected based upon the type of survey required and the anticipated types and levels of radioactivity at the location. Survey results are recorded and archived for future reference, reports, and radiation trend studies.

Area radiation surveys are performed on the basis of good ALARA practices and depend upon several factors such as location, actual or potential radiation levels, station status, and

occupancy factor. Surveys are periodically performed in areas subject to change. Examples of these areas inside the RCA include rooms with RHR piping during system operation, and areas outside the RCA include the Turbine Building condensate polishing skid area when there is primary to secondary leakage through one or more of the steam generators. The frequency of surveys may be weekly, monthly, quarterly, semi-annual, annual or more often as dictated by operating conditions and as directed by the Radiation Protection Manager. All survey results are recorded, filed and may be posted locally to assure adequate radiological controls. Caution placards are posted locally to comply with 10 CFR Part 20 requirements.

Contamination surveys are performed at selected locations throughout the station to evaluate the hazard due to removable and non-removable radioactive contamination. The selected locations represent strategic points where there exists the potential for the spread of contamination. Contamination surveys are made using the “smear”, “swipe”, or “large area masslin” technique or by using an appropriate portable instrument. Frequency of surveys depends upon factors such as the actual or potential radioactive concentration, occupancy factor, location, and station status. Routine surveys are performed weekly, monthly, quarterly, semi-annual, annual or more often as required by actual operating conditions and as directed by the Radiation Protection Manager. As required, contamination surveys are performed on personnel, equipment, and in uncontrolled areas to ascertain contamination levels and to ensure that control methods are adequate. Contamination survey results are recorded, filed, and as needed, posted for daily information.

Surveys to assess the airborne radioactivity levels are performed to ensure 10 CFR Part 20 limits are not exceeded, engineering controls are functioning, and respiratory protection techniques are adequate. Areas to be routinely surveyed are determined by a Radiation Protection Supervisor and are based upon the actual or potential hazard, task to be performed, and occupancy factor. Frequency of routine surveys for airborne radioactivity will vary depending upon several factors. For instance, selected areas may be surveyed daily, weekly, monthly or continuously depending upon the location, work to be performed, and actual or potential hazard. Monitoring may be performed with continuous air monitors (CAMs) or with grab samples.

In order to warn personnel of changing airborne conditions, CAM alarm setpoints are normally set at a value corresponding to 25% of the appropriate concentration values given in 10CFR20, Appendix B, Table 1, Column 3. These setpoints may be adjusted as necessary based on actual plant conditions in accordance with station procedures. Appropriate survey results necessary to demonstrate adequate radiological controls are recorded and filed and may be posted locally.

12.5.3.2 Radiological Work Control

Access to Radiologically Controlled Areas at CPNPP is controlled by administrative and physical measures and in accordance with procedural requirements. Station management assures entry control to radiation and high radiation areas through the administration of access permits that stipulate purpose of entry, work location, radiological conditions, surveillance and dosimetry requirements, protective clothing and equipment, and other procedural requirements. The objectives for issuing radiation work permits (RWP)/general access permits (GAP) is to provide the following:

1. Control the access to Radiologically Controlled Areas so that entrance is allowed only to those individuals whose job functions require entry.

2. Provide a detailed assessment of the actual and potential radiological hazards that are associated with the job function and area.

3. Ensure that proper protective measures are taken to safely perform the required duties in the area and to maintain occupational radiation exposure ALARA.

4. Ensure that the individual acknowledges his understanding of the radiological conditions, the protective and safety measures required, and his willingness to follow the GAP/RWP requirements.

5. Provide an avenue for dose-accounting that correlates the individual, the job function, and job for ALARA reports and studies.

6. Provides a communication check which ensures that the appropriate supervisors are aware of the task being performed, the radiological conditions, and the prescribed protective measures.

7. Provide a means for maintaining the accountability of personnel in RCAs.

General Access Permits (GAP) are issued for activities in RCAs where the radiation levels are known and the radiological conditions are not subject to erratic or large changes. The GAP requires administrative authorization and is issued in accordance with procedural requirements.

Radiation Work Permits (RWP) are issued for non-routine, specific activities in areas where radiation levels are significant and may change frequently, rapidly or significantly. Issuance, termination, authorization, protective measures, surveillance requirements and other stipulations for the RWP follow procedural requirements.

12.5.3.3 Posting of Radiological Conditions

Radiation areas, high radiation areas and other radiological zones are identified with the appropriate sign, label, or placard and the control of access into these areas is in accordance with the regulations of 10 CFR 20. Additionally, administrative controls for the posting of and access to high radiation areas are established in the plant Technical Specifications.

Areas that may be contaminated with radioactive material at CPNPP are decontaminated to a level that is reasonably achievable using the most effective methods and techniques. At times certain areas will be designated “contaminated areas” and therefore posted with the proper warning placards and barricaded to isolate the area from routine traffic patterns. Entry to these areas is controlled by Radiation Protection and allowed only through the issuance of a GAP or RWP. Personnel, equipment, and material exiting from contaminated areas are monitored by frisker or other appropriate radiation detecting instruments to prevent the spread of contamination to clean areas. Contaminated equipment is properly packaged and identified before removal from a contaminated area. A final survey will be conducted to ensure that all personnel, material, and equipment are free of significant contamination to provide assurance that no radioactive material is spread to unrestricted areas of the station.

12.5.3.4 Radiation Protection Training Program

The Manager, Nuclear Radiation Protection is responsible for the Radiation Protection Training Program at CPNPP. All permanent station personnel who are required to work in radiologically controlled areas (RCAs) complete the basic training courses, lectures, and exercises and demonstrate their proficiency and competence prior to being allowed to work in RCAs. The Radiation Protection Training Program maintains employee proficiency through periodic retraining lectures and exercises.

The degree of training received by employees is commensurate with the individual's job function and anticipated radiation hazards. The Radiation Protection Training Program includes as a minimum the following topics:

1. Fundamentals of radiation and radioactivity

2. Radiation Protection techniques

3. Biological effects of radiation

4. Measurement of radioactivity and radiation

5. Use of protective clothing and equipment

6. Radiation Protection procedures and instructions

7. Station and federal rules and regulations

8. Emergency planning

9. ALARA program - policy, concepts and methods

Visitors and temporary maintenance and service personnel will be trained in the above topics to the extent necessary to assure safe execution of their duties. As required, all non-permanent personnel and visitors are properly escorted unless they demonstrate an adequate knowledge of the appropriate topics included in the Radiation Protection Training Program. Additional training information is contained in Section 13.2.

12.5.3.5 Personnel Exposure Monitoring

Station and support personnel, who enter Radiologically Controlled Areas at CPNPP are monitored for external radiation exposure using Thermoluminescent Dosimeter (TLD) Badges, alarming electronic dosimeters or pocket ionization chambers (PICs). Issuance of TLD Badges, electronic dosimeters or PICs, is performed in accordance with approved station procedures. TLD Badges processing and dose determination are performed by an approved offsite dosimetry service organization with approved quality manual(s). Additional dosimetry equipment, such as multibadges is available and issued in accordance with procedural and radiological control requirements. The TLD Badge service organization shall maintain accreditation with the National Voluntary Laboratory Accreditation Program (NVLAP).

Electronic alarming dosimeters or direct reading pocket dosimeters are issued as a secondary method of determining gamma exposure for dose control/tracking purposes when TLD Badges are used as the primary means of monitoring exposure. The use, care and testing of pocket dosimeters follows the applicable guidance of Regulatory Guide 8.4 (February 1973).

Exposure records for each individual are maintained in accordance with Regulatory Guide 8.7 (June 1992). Current exposure data for each individual is available and may be reviewed by the individual upon request. Departmental exposure reports are issued to supervisors, upon request, to provide job scheduling assistance. Evaluation of employee exposure records by radiation protection personnel is performed to analyze individual and job exposure trends and to assist in maintaining occupational radiation exposures ALARA.

Internal radiation exposures are identified and assessed by the CPNPP Bioassay Program. This program follows the guidance of Regulatory Guide 8.9 (July 1993). Analysis of internal deposition of radioactive materials is performed using either a whole body counter or excreta samples. The whole body counter and personnel contamination monitor are used for in-vivo radiation measurements of station personnel, support personnel, and visitors. Those individuals who regularly enter areas where the potential exists for inhalation, ingestion, or absorption of radioactive materials are counted prior to initial entry. Special monitoring and evaluation of known or suspected uptakes is performed on a case-by-case basis in accordance with Regulatory Guide 8.9 (July 1993). If necessary, excreta samples are collected and sent to a qualified laboratory for analysis. Significant doses resulting from uptake of radioactive material(s) are calculated in accordance with ICRP 30 and documented on the individual's NRC Form 5 (equivalent) as specified in station procedures.

12.5.3.6 Respiratory Protection

It is management's intent to control airborne radioactivity levels as effectively as practicable by proper preventive measures, engineering controls, and good housekeeping techniques. At times, inadvertent airborne radioactivity problems will exist. Prompt assessment of the airborne activity levels is very important and was discussed previously in this section. Section 12.3.4 provides information on airborne radioactivity monitoring instrumentation.

Control of airborne radioactivity levels is assured through the use of the station's heating, ventilation and air conditioning (HVAC) systems and portable air movers and filters. The HVAC systems provide controlled air movement and filtration for those areas with a high potential for airborne radioactivity problems. As required, special control techniques are used, such as enclosures that isolate and vent airborne radioactivity arising from special work projects. To be used as a final alternative, respiratory protection equipment is available for use in those situations where airborne radioactivity hazards exist and other control measures prove inadequate.

The Manager, Nuclear Radiation Protection is the individual responsible for the development and implementation of the Respiratory Protection Program at CPNPP. The Respiratory Protection Program is developed through the guidance of Regulatory Guide 8.15 (October 1999) and satisfies the requirements of 10 CFR Part 20. A written policy statement exists to insure the provisions of the program are adequately followed and enforced. As a minimum the CPNPP Respiratory Protection Program will provide:

1. Written procedures covering such aspects of the program as description of the equipment, issuance of equipment, selection of equipment based upon hazards, maintenance of equipment and training.

2. Training and fitting sessions for those individuals whose job may entail the wearing of respiratory protection equipment.

3. Proper storage facilities, maintenance, and quality control for equipment used in the respiratory protection program.

4. An adequate medical surveillance program for respirator users.

5. Periodic inspection and surveillance of routine and emergency respiratory equipment.

6. Use of only NIOSH/MSHA-certified or NRC-authorized equipment.

7. A periodic review of the overall Respiratory Protection Program, based upon inspection, maintenance, and quality control reports and bioassay results, to verify the effectiveness of the program.

12.5.3.7 Radioactive Materials Control

Procedures and instructions are developed to control, handle, and store radioactive materials, including radioactive sources and contaminated materials such as tools and equipment. The objectives of these procedures and instructions are to maintain accountability of material, to satisfy regulatory requirements, and to ensure that occupational radiation exposures due to this material are ALARA. Subjects covered by these procedures and instructions include, but are not limited to:

1. Properly identifying radioactive material as required by applicable regulatory and procedural requirements. Warning labels and signs are applied to areas and containers.

2. Utilizing adequate storage facilities for radioactive materials. Storage areas for licensed sealed sources are maintained locked, except when in use. Access to radioactive material is controlled (e.g., by RWP, shielded or locked if appropriate) to minimize exposure to personnel.

3. Using the elements of time, distance, and shielding to work safely with this material and to maintain exposures ALARA. Special extension and remote handling tools are used when applicable.

4. Periodically testing licensed radioactive material sealed sources to verify the integrity of the sealed material.

5. Inventorying non-exempt (reference 10 CFR 30) radioactive sources at regular intervals to maintain accountability.

6. Instructions for proper actions to be taken in the event of leakage and spills.

7. Requirements for the movement and use of radioactive materials including shielding requirements, area control, and accountability requirements.

TABLE 12.5-1
RADIATION MONITORING INSTRUMENTATION AND OTHER RADIATION PROTECTION EQUIPMENT

(Sheet 1 of 2)

The minimum quantity, type of radiation sensitivity, typical range of instruments and equipment are shown below:

	Quantity (Approximate)	Type of Radiation Detected	Range (Typical)
SURVEY INSTRUMENTS:	10 per unit	Beta, Gamma	0-50 R/hr & 0-5R/hr
	15 per unit	Gamma	0-5 R/hr
	4 per unit	Beta, Gamma	0-200 R/hr
	1 per unit	Gamma	0-10,000 R/hr
	1 per unit	Neutron	0-5 Rem/hr
	1 per unit	Alpha	0-20,000 dpm
ELECTRONIC ALARMING DOSIMETERS:	200 per unit	Gamma	1 mR - 100R
POCKET ION CHAMBER DOSIMETERS:	150 per unit	Gamma	0-200 mR or 0-500 mR
(NOTE: Direct reading pocket ion chambers (PICs) in the ranges indicated may be used in lieu of electronic alarming dosimeters)	50 per unit	Gamma	0-1.5 R

TABLE 12.5-1
RADIATION MONITORING INSTRUMENTATION AND OTHER RADIATION PROTECTION EQUIPMENT

(Sheet 2 of 2)

	Quantity (Approximate)	Type of Radiation Detected	Use
OTHER AVAILABLE EQUIPMENT:			
Gamma Spectroscopy System	1	Gamma	Routine counting of liquids, gases and solids
Whole Body Counting System	2	Gamma	Used for in Vivo bioassay
Gas Proportional Counter	1	Beta, Gamma, Alpha	Used for counting smear and air samples
Liquid Scintillation Counter	1	Beta, Gamma	Routine tritium and gross beta analysis
Personnel Contamination/ Portal Monitor	2 per unit	Beta, Gamma	Personnel contamination monitoring at RCA exits. Personnel contamination monitors are also used to perform passive whole body counting (in-vivo bioassay).
Continuous Air Monitor (CAMs)	2 per unit	Beta	Work area air monitoring
Airborne Grab Sampler	15 per unit	Beta Gamma	Work area air monitoring
Count Rate Meter	25 per unit	Beta	Contamination frisking

TABLE 12.5-2
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13.0 CONDUCT OF OPERATIONS

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13.1 ORGANIZATIONAL STRUCTURE OF APPLICANTS

Luminant Power Company is a subsidiary of Energy Future Holdings Corporation (EFH). A simplified organizational diagram for Luminant Power Company in the EFH Corporate structure is shown in Figure 13.1-1.

Luminant Power has corporate responsibility for the design, construction and operation of CPNPP, which includes the functions of procurement, fuel management and quality assurance. Within Luminant Power, the fossil group is responsible for the operation and related activities associated with the fossil fueled plants. The nuclear group, designated as the Nuclear Generation Group, has been delegated to furnish design, engineering, construction, licensing, operation and fuel management support to CPNPP. The Nuclear Generation Group has been split into the following organizations: Nuclear Engineering & Support, Nuclear Operations, Nuclear Oversight, External Affairs, and Strategic Support.

13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

Project management and technical support for CPNPP are the responsibility of Luminant Power.

13.1.1.1 Design and Operating Responsibilities

The design and operating responsibilities are divided into three categories: design and construction activities, pre-operational activities and technical support for operation. The Nuclear Generation organizations are responsible for these activities as discussed below.

13.1.1.1.1 This section deleted in its entirety as it does not apply to an operating plant.

13.1.1.1.2 This section deleted in its entirety as it does not apply to an operating plant.

13.1.1.1.3 Technical Support for Nuclear Operations

Technical services and backup support for Nuclear Operations are furnished by Nuclear Engineering. Personnel are available who are competent in technical matters related to plant safety and other engineering and scientific support aspects. In the event Nuclear Operations needs assistance with specific problems, the services of qualified individuals will be engaged as appropriate.

The special capabilities that are available are:

- Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical and materials, chemistry, and instruments and controls engineering
- Nuclear Safety
- Plant Chemistry
- Health Physics
- Fueling and refueling operations support

- Maintenance support
- Licensing
- Industrial Safety
- Fire Protection

13.1.1.2 Organizational Arrangement

Responsible positions in the organization are described below. Certain executive and management positions may have deputies assigned to the position. Deputies may act with the full authority of the position to which assigned.

CEO, Luminant Power - The CEO, Luminant Power has the overall corporate responsibility for the design, construction and operation of CPNPP. He provides guidance to the Senior Vice President & Chief Nuclear Officer. The Nuclear Generation Group is shown in Figure 13.1-2.

Senior Vice President & Chief Nuclear Officer- The responsibilities of the Senior Vice President & Chief Nuclear Officer are:

- Directing the engineering, construction, start-up, testing, operation and maintenance of CPNPP.
- Providing the Quality Assurance Program and associated audit services applicable to all nuclear activities.
- Providing engineering services, technical services, nuclear fuel services, and licensing services to Nuclear Operations.
- Providing technical direction and guidance to the Nuclear Generation Vice Presidents and other direct reporting managers.
- Establishing and maintaining the Operations Review Committee.
- Providing corporate responsibility for overall plant nuclear safety and taking any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.
- Maintaining the independence and credibility of the Program.
- Ensuring adequate responses are provided to employees with concerns.

13.1.1.2.1 Organization - Nuclear Operations

The Site Vice President reports directly to the Senior Vice President & Chief Nuclear Officer.

The responsibilities of the Site Vice President are:

- Ensuring CPNPP operation, maintenance, emergency planning, radiation protection, nuclear training and plant support activities are conducted in compliance with federal, state, and local laws, regulations, licenses, codes, and within established corporate and Nuclear Generation policies, plans and procedures.

- Providing direction and guidance to the Plant Manager; Director, Organizational Effectiveness; and Manager, Environmental Services.

- Ensuring the safe and reliable operation of the plant and overall responsibility for security of the plant.

The CPNPP operating organization is discussed in Section 13.1.2 and illustrated in Figure 13.1-3. Organizations which support plant operations are discussed below.

Plant Manager - The Plant Manager reports directly to the Site Vice President and is responsible for:

- Operational and maintenance support of CPNPP.

- Management of all operations activities at CPNPP.

- Providing direction and guidance to the Director, Operations; Director, Maintenance; Manager, Nuclear Radiation Protection, and Director, Work Management.

Director, Operations - The Director, Operations reports directly to the Plant Manager and is responsible for:

- Operations of CPNPP.

- Management and training of Operations Department personnel.

- Coordinating the generation of power and changes in operating modes.

- Participating in power ascension test program and refueling efforts.

Director, Maintenance - The Director, Maintenance reports directly to the Plant Manager and is responsible for:

- Maintenance activities associated with mechanical and electric equipment, instrumentation and controls.

- Fire Protection.

- Implementing the preventative maintenance activities during routine operation and refueling outages and activities associated with the power ascension test program are conducted in accordance with approved procedures and instructions, regulatory requirements and applicable policies and directives.

Manager, Nuclear Radiation Protection - The Manager, Nuclear Radiation Protection reports directly to the Plant Manager and is responsible for:

- CPNPP Radiation Protection program.
- Supervision of Radiation Protection Manager and Supervisors.
- Transportation of radioactive material.
- Implementation of station policy of maintaining operational radiation exposures “as low as reasonable achievable”.
- Safety Services.

Director, Organizational Effectiveness - The Director, Organizational Effectiveness reports directly to the Site Vice President and is responsible for:

- Overseeing the training of operations, maintenance and support personnel of the Comanche Peak Nuclear Power Plant, or CPNPP, to ensure safe, reliable and cost-effective operation.
- Monitoring and improving personnel and plant performance to ensure that CPNPP is operated and maintained in accordance with the provisions of the operating license, Nuclear Regulatory Commission and other federal regulations including the state of Texas regulations and permits, industry codes and standards, and Luminant requirements.

Director, Nuclear Training - The Director, Nuclear Training reports directly to the Director, Organizational Effectiveness and is responsible for:

- Directing the analysis, design, development, implementation, evaluation and revision of nuclear training programs in order to provide personnel with the requisite skills and knowledge for effectively performing functions important to the operation and maintenance of CPNPP.

Director, Nuclear Performance Improvement - The Director, Performance Improvement reports directly to the Director, Organizational Effectiveness and is responsible for:

- Trending and analysis of conditions adverse to quality.
- Reviewing and assessing nuclear industry operating experience impact on CPNPP.
- Administering and facilitating the Corrective Action Program.
- Administering and facilitating the Human Performance Program.
- Administering and facilitating the Self Assessment and Benchmarking Programs.

Director, Work Management - The Director, Work Management reports directly to Plant Manager and is responsible for:

- CPNPP work control program.

- Planning, maintenance and work order packages.
- Providing CPNPP an integrated schedule.
- Outage management.

13.1.1.2.2 Organization - Nuclear Engineering & Support

Vice President, Nuclear Engineering and Support (Vacant) - The responsibilities of the Vice President, Nuclear Engineering and Support are:

- Total engineering responsibility for the Nuclear Engineering Organization including Project Development, Plant Modification, Reactor Engineering, Technical Support, and System Engineering.
- Represent Luminant Power in committees and task forces established for the nuclear industry.
- Providing direction and guidance to the Director, Site Engineering; Director, Engineering Support; Manager, Project Engineering, and Manager, Technical Support.

Director, Site Engineering - The Director, Site Engineering reports directly to the Senior Vice President and Chief Nuclear Officer and is responsible for:

- Implementing a system health program to manage system performance problems and recommend effective improvements; implementing the predictive maintenance programs; implementing the maintenance effectiveness monitoring program (Maintenance Rule Program); and performing assigned surveillance and preventive maintenance test procedures.
- Providing support to Nuclear Generation and to other Luminant Power organizations in the areas of Nuclear Fuel Management, Reload Design Engineering, Thermal Hydraulic Analysis, Integrated Risk and Availability Modeling, Nuclear Safety Analysis, Severe Accident Management, and Core Performance.

Director, Project Engineering and Support - The Director, Project Engineering and Support reports directly to the Senior Vice President and Chief Nuclear Officer and is responsible for:

- Providing for the development and implementation of specific major scope plant modifications and/or engineering projects including interface with involved offsite vendor organizations.
- Providing for the development and implementation of an integrated administrative services program.

Manager, Technical Support - The Manager, Technical Support reports directly to the Director, Site Engineering and is responsible for:

Assuring the consistency of design documentation; providing Operations with timely design engineering services for analyses and technical evaluations; assuring that design activities

conducted for Comanche Peak meet the requirements of the design control program; assuring that design outputs are consistent with the design basis of the plant; and providing engineering specialists.

13.1.1.2.3 Organization - Regulatory Affairs

Manager, Regulatory Affairs - The Manager, Regulatory Affairs reports directly to the Senior Vice President & Chief Nuclear Officer and is responsible for:

- Providing liaison with governmental agencies and implementing the actions necessary to obtain and maintain the permits, licenses, and approvals needed to construct and operate CPNPP.

- Providing technical expertise to evaluate and resolve issues which affect permits and licenses.

- Assisting corporate management in the development and implementation of positions on licensing issues consistent with Nuclear Generation Group policies and procedures.

- Providing formal communication to regulatory agencies (e.g., NRC) with any control over CPNPP, except for that communication assigned to Environmental Services.

13.1.1.2.4 Organization - Nuclear Oversight

Manager, Nuclear Oversight - The Manager, Nuclear Oversight reports to the Senior Vice President & Chief Nuclear Officer and is responsible for:

- Ensuring that activities at CPNPP are conducted in a manner that is conducive to quality results.

- Keeping the Senior Vice President & Chief Nuclear Officer apprised of the status of Nuclear Oversight activities.

- Recommending actions or methods to improve or correct conditions which are adverse to quality.

- Developing, implementing and managing the vendor control program which includes the review of vendor QA programs, review of procurement documents, evaluation of vendor activities, and source and/or receipt inspection.

- Directing site related Qualify Control activities for CPNPP Operations and Maintenance.

- Performing observations of plant activities to ensure the activities are performed safely, and to ensure plant safety. Make independent safety recommendations to the Senior Vice President & Chief Nuclear Officer on ways to improve the overall quality and safety of operation, maintenance, and engineering activities.

- Ensuring that Quality Assurance Program requirements are met by conducting evaluations (i.e., audits, surveillances, assessments, inspections...).

- Providing for independent verification of critical attributes associated with safety-related equipment or work activities.

- Providing, when necessary, independent review and concurrence for quality-related activities such as procurement, nonconformance and corrective action, stop-work orders, and other activities as designated in the Quality Assurance Program.

- Performing systematic monitoring and assessment of plant operation and maintenance activities, and provide senior management with critical assessments of plant practices and issues that may affect nuclear safety and the operation and maintenance of CPNPP.

- Providing engineering services to support the Operations Review Committee (ORC)

- Providing Health Physics review for those programs necessary for radiological protection of Company personnel and property, the environment and the general public.

- Supporting the development of appropriate emergency plans for Luminant Power in accordance with regulatory guidelines and requirements, and performance of the 10CFR50.54(t) independent review of the Emergency Planning Program.

Manager, SAFETEAM - The Manager, Safeteam reports to the Manager, Nuclear Oversight and is responsible for:

- Managing the Safeteam Program for the review and investigation of employee safety concerns.

- Ensuring both departing employees and employees with concerns are interviewed.

13.1.1.3 Qualification of Management and Technical Support Personnel

Resumes of current key Nuclear Generation managerial and technical support personnel are provided in Appendix 13.1A.

13.1.2 OPERATING ORGANIZATION

13.1.2.1 Nuclear Operations Organization

The Plant Manager is responsible for the operation and maintenance of the Comanche Peak Nuclear Power Plant (CPNPP). Reporting to the Plant Manager are the Director, Operations; the Director, Maintenance; Director, Work Management and the Manager, Nuclear Radiation Protection. Reporting interfaces and relationships are shown on Figures 13.1-2 and 13.1-3. Nuclear Oversight is illustrated on Figure 13.1-2 and discussed in Section 13.1.1.2.4 (organization) and Chapter 17 (program).

13.1.2.1.1 Personnel Functions, Responsibilities, and Authorities

Personnel responsibilities relating to membership on the Station Operations Review Committee (SORC) and the Operations Review Committee (ORC) are discussed in Section 13.4.

13.1.2.1.2 Operations Department

The Director, Operations is responsible for the operation of CPNPP and the management and training of Operations Department personnel. He coordinates the generation of power and changes in plant operating modes and participates in the power ascension test program and the refueling efforts.

Reporting to the Director, Operations are the Shift Operations Manager, the Manager, Nuclear Operations Support, and the Manager, Nuclear Chemistry. The Director, Operations provides technical assistance for the development and maintenance of Operations Department procedures to ensure CPNPP is operated as prescribed in Section 13.5. He is also responsible for the operation of the radioactive waste handling systems, for the processing and packaging of radioactive waste.

The Shift Operations Manager is responsible for post-trip reviews, for refueling support and for reactor operator training support. The Shift Operations Manager directs the Shift Managers and is responsible for ensuring that shift operations personnel are trained and qualified (see Section 13.2). The Shift Operations Manager is the position designated to meet ANSI N18.1-1971 qualification requirements for “Operations Manager” and is required to maintain a USNRC Senior Reactor Operator License.

Shift Operations

The Shift Managers are members of management responsible to the Shift Operations Manager for the operation of the CPNPP units. Each Shift Manager is responsible for supervising the evolutions conducted during his shift and ensuring that they are conducted in accordance with the operating license, station procedures, and applicable directives and policies. The Shift Managers are responsible for supervising shift operations personnel and for conducting on-shift training. During periods when senior management personnel are not on site, the Shift Managers assume responsibility for all station activities. The Shift Managers are required to maintain an USNRC Senior Reactor Operator License.

The Unit Supervisors are members of management and assist the Shift Managers in discharging their responsibilities for supervision of the CPNPP units. The Unit Supervisor may assume the duties of the Shift Manager in his absence. The Unit Supervisors are required to maintain an USNRC Senior Reactor Operator License.

The Reactor Operators are responsible for routine evolutions on their assigned unit and for monitoring the status of that unit. The Reactor Operators are supervised by the Shift Manager or a Unit Supervisor and are required to maintain a USNRC Reactor Operator License.

Plant Equipment Operators work under the direction of the Shift Manager, Unit Supervisor, or Radwaste Supervisor. The Plant Equipment Operators’ responsibilities include operating equipment from the Control Room and operating and servicing equipment remote from the Control Room at the direction of Control Room operations personnel.

Shift Technical Advisors will be on each shift unless the Shift Manager or the individual with a Senior Reactor Operator license meets the qualifications described in Option 1 of the Commission Policy Statement on Engineering Expertise (50 FR 43621, October 28, 1985). They will report to the Shift Managers.

A Radwaste Supervisor is responsible to the Manager, Operations Support for the operation of the liquid and gaseous waste systems. He supervises the various liquid and gaseous waste processing activities and serves as an interface between the Operations Department and other CPNPP personnel involved in waste or water handling evolutions.

Chemistry

The Manager, Nuclear Chemistry is responsible for the supervision of chemistry personnel, for monitoring and maintaining the station's fluid systems chemistry. In discharging these responsibilities, he ensures that his personnel are trained and that safety-related activities are conducted in accordance with applicable procedures, instructions, policies and regulations.

13.1.2.1.3 Maintenance Department

The Director, Maintenance is responsible for maintenance activities associated with mechanical and electrical equipment, instrumentation and controls, and for implementing the preventive maintenance program. The Director, Maintenance ensures that maintenance personnel are trained and qualified. He ensures that maintenance activities during routine operation and refueling outages and activities associated with the power ascension test program are conducted in accordance with approved procedures and instructions, regulatory requirements, and applicable policies and directives. The Director, Maintenance is responsible for developing and maintaining procedures and instructions as described in Section 13.5.

Maintenance Teams

The Maintenance Teams are divided into one multi-discipline team (Electrical and Mechanical) and one Instrumentation and Control team. These teams are system oriented established to provide ownership and accountability within the maintenance organization.

The Maintenance Team Managers are responsible to the Director, Maintenance for the maintenance of electrical and mechanical plant systems and their instrumentation and control systems. They ensure that the electricians, mechanics, and I&C technicians are trained and that safety-related activities are conducted in accordance with applicable procedures, instructions, policies and regulations. They are responsible for managing their respective areas/systems through the Maintenance Team Supervisors who direct the day-to-day activities of their personnel.

Maintenance Plant Support Team

The Maintenance Plant Support Manager is responsible to the Director, Maintenance for maintaining plant valves, welding and machining activities, and oversight to the PROMPT Team.

PROMPT Team

The PROMPT Team is a 24 hour/365 day multi-discipline maintenance team reporting to the Maintenance Plant Support Manager. Specific duties and responsibilities include, but are not limited to, ensuring PROMPT Team activities are performed in accordance with the applicable site procedures. The PROMPT Team is also responsible for providing immediate response to plant emergent maintenance items.

Maintenance Support Team

The Maintenance Manager is responsible to the Director, Maintenance for providing technical, administrative and field support for the Maintenance Department.

13.1.2.1.4 Radiation Protection and Safety Services

Radiation Protection

The Manager, Nuclear Radiation Protection is responsible for the supervision of the Radiation Protection Supervisors, for the transportation of radioactive material, for the CPNPP Radiation Protection program (see Section 12.5) and for implementation of the station policy of maintaining operational radiation exposures "as low as reasonably achievable." In discharging these responsibilities, he ensures that his personnel are trained and that safety-related activities are conducted in accordance with applicable procedures, instructions, policies and regulations.

Safety Services

The Industrial Safety Manager is responsible for industrial safety.

13.1.2.1.5 Plant Support

The Manager, Work Control/Outage is responsible for planning and scheduling and outage management.

Planning and Scheduling

The Manager, Work Control/Outage is responsible for:

- maintaining the work control process, scheduling on-line work and tests, and administrating the risk assessment process.

- preparation and execution of planned outages, scheduling outage activities, ensuring the implementation of the risk assessment process on outage activities, and incorporation of outage lessons learned.

Station Security

The Manager, Nuclear Security is responsible for the overall development and implementation of the security program at CPNPP as outlined in the Security plans.

Emergency Planning

The Manager, Nuclear Emergency Planning is responsible for the development of the Emergency Plan and procedures, maintenance of emergency response facilities and equipment, and training of the emergency response organization. The Manager, Nuclear Emergency Planning is also responsible for interfacing with local, state and federal officials to ensure integrated onsite and offsite plans.

13.1.2.2 Supervisory Succession

The Plant Manager is responsible for the operation of CPNPP. If the Plant Manager is absent or becomes incapacitated, then, unless otherwise designated, the following station staff assume the subject responsibilities in the order listed:

1. Director, Operations

2. Director, Maintenance

During back shift and weekend periods when the station staff is not on site, the Shift Manager is responsible for all activities at CPNPP.

13.1.2.3 Shift Crew Composition

The minimum on-duty shift complement for various modes of single and dual unit operation is shown in Table 13.1-2 and is as follows:

Two USNRC Licensed Operators should be in the Control Room for each reactor while undergoing a start-up, scheduled shutdown or reactor trip recovery.

With two units licensed to operate (both Units in Mode 5 or 6), each shift crew shall have at least six members, including one Shift Manager and two USNRC Licensed Operators.

With two units licensed to operate and one or both operating (Mode 1, 2, 3, 4), each crew shall have at least eight members, including one Shift Manager, one Unit Supervisor and three USNRC Licensed Operators.

An organization chart for CPNPP for one and two unit operation is provided in Figure 13.1-3.

13.1.3 QUALIFICATION REQUIREMENTS FOR PLANT PERSONNEL

13.1.3.1 Minimum Qualification Requirements

The minimum qualification requirements for Licensed Senior Reactor Operators and Reactor Operators are in accordance with Regulatory Guide 1.8 Rev. 3 and are displayed in Table 13.1-1. All other plant personnel will meet or exceed the minimum qualification requirements of Regulatory Guide 1.8, Rev. 2.

13.1.3.2 Qualifications of Plant Personnel

Resumes of current key operating organization personnel are provided in Appendix 13.1A.

TABLE 13.1-1
MINIMUM QUALIFICATIONS OF PLANT PERSONNEL IN ACCORDANCE WITH REGULATORY GUIDE 1.8, REVISION 2

(Sheet 1 of 4)

Notes: (1) Recommended, but not required (2) Recommended in addition to experience (3) Experience required to independently perform job (4) In accordance with Regulatory Guide 1.8, Rev. 3, May 2000	H. S. DIPLOMA	B.S. in ENGINEERING OR SCIENCE	POWER PLANT EXPERIENCE	NUCLEAR PLANT EXPERIENCE	EXPERIENCE IN AREA OF EXPERTISE	EQUIVALENT EDUCATIONAL EXPERIENCE	REACTOR OPERATOR LICENSE REQUIRED	SENIOR REACTOR OPERATOR LIC/CERT REQUIRED
SITE VICE PRESIDENT OR PLANT MANAGER		X	10	3		4		X(1)
DIRECTOR, OPERATIONS		X(1)	8	3		2		X(1)
DIRECTOR, MAINTENANCE		X(1)	7	1		2		
MAINTENANCE DEPARTMENT MANAGERS		X(1)	5	1		4		
SHIFT OPERATIONS MANAGER		X(1)	8	3		2		X
SHIFT MANAGER	X		3(4)	3(4)		2(2)		X
SENIOR REACTOR OPERATOR	X		3(4)	3(4)		2(2)		X

TABLE 13.1-1
MINIMUM QUALIFICATIONS OF PLANT PERSONNEL IN ACCORDANCE WITH REGULATORY GUIDE 1.8, REVISION 2

(Sheet 2 of 4)

Notes: (1) Recommended, but not required (2) Recommended in addition to experience (3) Experience required to independently perform job (4) In accordance with Regulatory Guide 1.8, Rev. 3, May 2000	H. S. DIPLOMA	B.S. in ENGINEERING OR SCIENCE	POWER PLANT EXPERIENCE	NUCLEAR PLANT EXPERIENCE	EXPERIENCE IN AREA OF EXPERTISE	EQUIVALENT EDUCATIONAL EXPERIENCE	REACTOR OPERATOR LICENSE REQUIRED	SENIOR REACTOR OPERATOR LIC/CERT REQUIRED
SHIFT TECHNICAL ADVISOR		X		1				
REACTOR OPERATOR	X		2(4)	1			X	
NUCLEAR EQUIPMENT OPERATOR	X							
MANAGER, CORE PERFORMANCE ENGINEERING		X		2(3)				
MECHANIC					3(3)			
ELECTRICIAN					3(3)			
INSTRUMENT & CONTROL TECHNICIAN					2(3)	1(2)		

TABLE 13.1-1
MINIMUM QUALIFICATIONS OF PLANT PERSONNEL IN ACCORDANCE WITH REGULATORY GUIDE 1.8, REVISION 2

(Sheet 3 of 4)

Notes: (1) Recommended, but not required (2) Recommended in addition to experience (3) Experience required to independently perform job (4) In accordance with Regulatory Guide 1.8, Rev. 3, May 2000	H. S. DIPLOMA	B.S. in ENGINEERING OR SCIENCE	POWER PLANT EXPERIENCE	NUCLEAR PLANT EXPERIENCE	EXPERIENCE IN AREA OF EXPERTISE	EQUIVALENT EDUCATIONAL EXPERIENCE	REACTOR OPERATOR LICENSE REQUIRED	SENIOR REACTOR OPERATOR LIC/CERT REQUIRED
DIRECTOR, WORK MANAGEMENT		X(1)	5	1		2		
DIRECTOR, SITE ENGINEERING		X	8	1		4		
MANAGER, TECHNICAL SUPPORT		X	8	1		4		
DIRECTOR, PROJECT ENGINEERING AND SUPPORT		X	8	1		4		
MANAGER, NUCLEAR RADIATION PROTECTION		X(1)		3	4	4		
MANAGER, NUCLEAR CHEMISTRY		X(1)		1	5	4		
STAFF HEALTH PHYSICIST		X(1)						

TABLE 13.1-1
MINIMUM QUALIFICATIONS OF PLANT PERSONNEL IN ACCORDANCE WITH REGULATORY GUIDE 1.8, REVISION 2

(Sheet 4 of 4)

Notes: (1) Recommended, but not required (2) Recommended in addition to experience (3) Experience required to independently perform job (4) In accordance with Regulatory Guide 1.8, Rev. 3, May 2000	H. S. DIPLOMA	B.S. in ENGINEERING OR SCIENCE	POWER PLANT EXPERIENCE	NUCLEAR PLANT EXPERIENCE	EXPERIENCE IN AREA OF EXPERTISE	EQUIVALENT EDUCATIONAL EXPERIENCE	REACTOR OPERATOR LICENSE REQUIRED	SENIOR REACTOR OPERATOR LIC/CERT REQUIRED
STAFF CHEMIST		X(1)						
RADIATION PROTECTION TECHNICIAN					2(3)	1(2)		
CHEMISTRY TECHNICIAN					2(3)	1(2)		
MANAGER, REGULATORY AFFAIRS	X	X(1)	5	4	6(3)	4		
MANAGER, NUCLEAR OVERSIGHT	X			2	6(3)	4		

TABLE 13.1-2
MINIMUM SHIFT CREW COMPOSITION[a]

(Dual Units with common Control Room)

POSITION[c]	MINIMUM CREW COMPOSITION[b]		
	BOTH UNITS in MODES 1,2,3, or 4	ONE UNIT in MODE 1,2,3, or 4; and ONE UNIT in MODE 5,6, or DEFUELED	BOTH UNITS in MODE 5,6, or DEFUELED
SM	1	1	1
SRO	1	1	NONE
STA	1	1	1
RO	3	3	2
PEO	4	4	4
RP Tech	2	2	2
Chem Tech	1	1	1

a) Additional minimum on-shift staffing requirements are contained in the CPNPP Emergency Plan.

b) As allowed by note 1 in 10 CFR 50.54(m)(2)(i), temporary deviations from the numbers of licensed operators required by this table shall be in accordance with the following criteria: The minimum Operations shift crew composition may be one less than shown in Table 13.1-2 for not more than (2) hours to accommodate unexpected absences of on-duty crew members, provided immediate action is taken to restore the crew composition within the minimum shown in Table 13.1-2. This exception does not permit any crew composition to be unmanned upon shift turnover due to an oncoming crew member being late or absent.

c) SM - (Shift Manager) with a dual unit Senior Reactor Operator license assigned to both units when either unit contains fuel. During the absence of the SM from the control room, any currently licensed SRO shall be designated to assume the control room command function.

SRO - Individual with a dual unit Senior Reactor Operator license assigned to each unit containing fuel. During core alterations on either unit, at least one currently licensed SRO (or SRO limited to fuel handling) will be present and responsible for fuel handling activities with no other concurrent duties assigned.

RO - Individual with a dual unit Reactor Operator or Senior Reactor Operator license assigned to each unit containing fuel. At least (1) RO will be assigned as a relief operator when either unit is in MODE 1, 2, 3, or 4.

PEO - A plant equipment operator shall be assigned to each unit containing fuel.

STA - (Shift Technical Advisor) to be assigned to both units in all MODES or when DEFUELED. The STA position may be filled by an on-shift SRO provided the individual meets the dual role requirements described in the Commission Policy Statement on Engineering Expertise on Shift (50 FR 43621) and has dose assessment capability. The SM shall not fulfill the duties as Emergency Coordinator and dose assessor concurrently.

RP/Chem Tech -

At least (2) Radiation Protection Technicians and (1) Chemistry Technician shall be on-site to assume emergency response functions. The Radiation Protection and the Chemistry Technicians may be less than the minimum requirements for a period of 2 hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

APPENDIX 13.1A - RESUMES OF THE FOLLOWING KEY LUMINANT POWER CPNPP PERSONNEL ARE PROVIDED:

Thomas P. McCool	Site Vice President, Acting
Kenneth J. Peters	Senior Vice President & Chief Nuclear Officer, Acting
Al Marzloff	Shift Operations Manager
David A. Goodwin	Director, Work Management
John R. Dreyfuss	Plant Manager
Timothy A. Hope	Manager, Regulatory Affairs
Donna Christiansen	Director, Nuclear Training
Doyce W. McGaughey	Director, Operations
Michael G. Stakes	Director, Maintenance
Steven K. Sewell	Director, Organizational Effectiveness
Deborah Farnsworth	Director, Nuclear Performance Improvement
James L. Patton	Manager, Nuclear Oversight
Robert Deppi	Director, Engineering Projects and Support
John Taylor	Director, Site Engineering
Deborah O'Connor	Manager, Nuclear Radiation Protection

Thomas P. McCool - Site Vice President, Acting

Education:

1985 Maine Maritime Academy

B.S., Marine Engineering

Minor in Nuclear Engineering and Engineering Science

2004 Leadership Academy (AEP)

2006 Exelon Advanced Management Program (AMP)

2008 INPO Senior Nuclear Plant Manager Course

2012 Certified Facilitative Leadership Instructor

Experience:

1985 -	General Foods Corporation Co-gen Plant Operator
1986 -	Maine Yankee Atomic Nuclear Plant Operator
1987 -	South Texas Nuclear Plant Reactor Operator
1990 -	Waterford Steam Electric Unit 3 Senior Training Instructor
1995 -	Beaver Valley Power Station Assistant Nuclear Shift Supervisor
1999 -	DC Cook Nuclear Plant Assistant Production Manager (OPS Superintendent)
2005 -	Exelon Nuclear Corporate Training Director
2007 -	Exelon Nuclear Operations Director
2010 -	Exelon Nuclear Maintenance Director
2010 -	Plant Manager, San Onofre Nuclear Generating Station
2013 -	Vice President, Station Support CPNPP
2014 -	Vice President Nuclear Engineering & Support CPNPP
2016 -	Site Vice President, Acting

Kenneth J. Peters – Senior Vice President and Chief Nuclear Officer, Acting

Education/Training:

Mentor, INPO Senior Nuclear Plant Management course

Participant, INPO Senior Nuclear Plant Management course

Senior Reactor Operator Certification, Indian Point

B.S., Electrical Engineering, NJIT

Experience:

1989 - 2006	Waterford 3 and Indian Point 3
	Director of Safety Assurance
	Design Engineering Manager
	Corrective Action & Assessment Manager
	Shift Outage Manager (temporary rotations) 1997 to 2001
	Licensing Manager
2006 - 2012	Pacific Gas and Electric company - Diablo Canyon Nuclear Power Plant
2006	Sr. Engineering Director/Engineering Director
2008	Station Director
2011	VP Engineering & Projects
2012	Luminant - Comanche Peak Nuclear Power Plant
	Site Vice President
2016 -	Senior Vice President and Chief Nuclear Officer, Acting

Alan Marzloff - Shift Operations Manager

Education:

B. S. Nuclear Engineering Technology - Thomas Edison State College, 2006

Experience:

1990 -	Operator, Trojan Nuclear Power Plant, Portland General Electric
1993 -	Nuclear Equipment Operator, CPNPP
2003 -	Reactor Operator, CPNPP
2008 -	Unit Supervisor, CPNPP
2012 -	Shift Manager, CPNPP
2015 -	Shift Operations Manager, CPNPP

David A. Goodwin - Director, Work Management

Education:

B.S., Nuclear Engineering, University of Virginia, 1982

B.S., Electrical Engineering, University of Virginia, 1982

Experience:

1980 -	Licensed as Reactor Operator, University of Virginia Research Reactor.
1982 -	Employed by TU Electric as a Shift Technical Advisor, CPSES.
1984 -	Licensed as Senior Reactor Operator, CPSES.
1989 -	Assigned as a Unit Supervisor/Shift Technical Advisor in the Shift Operations Department, CPSES.
1991 -	Assigned as Shift Manager in the Shift Operations Department, CPSES.
1996 -	Assigned as Operations Support Manager, CPSES.
1999 -	Assigned as Acting Shift Operations Manager, CPSES.
1999 -	Assigned as SMART Team Manager in Maintenance, CPSES.
2007 -	Assigned as Shift Operations Manager, CPSES.
2008 -	Assigned as Director, Operations, CPSES.
2009 -	Assigned as Director, Engineering Projects, CPNPP.
2013 -	Assigned as Director, Work Management, CPNPP

John R. Dreyfuss - Plant Manager

Education:

B. S. Nuclear Engineering - Penn State University

SRO License, Nine Mile Point Unit 1, Docket 55-62243

SRO Certifications: Vermont Yankee, Nine Mile Point, Shoreham

INPO Senior Nuclear Plant Manager Course - 2001

Experience:

1983 -	E. G. Todd Associates, Account Executive
1985 -	Long Island Lighting Company, Shoreham Nuclear Power Station
1990 -	Niagara Mohawk Power Corporation, Nine Mile Point
1999 -	Technical Services Manager, VYNPC/Vermont Yankee
2000 -	Manager Systems, Programs and Components Engineering, Vermont Yankee
2003 -	Manager Engineering Support, Vermont Yankee
2004 -	Director Engineering, Vermont Yankee
2006 -	Director Nuclear Safety Assurance, Vermont Yankee
2010 -	Assistant to Entergy Nuclear North Chief Operations Officer, White Plains, NY
2012 -	General Manager Plant Operations, Pilgrim Nuclear Power Station
2012 -	Director Fukushima Project, Entergy Corporate
2013 -	Director, Organizational Effectiveness, CPNPP
2014 -	Plant Manager

Timothy A. Hope - Manager, Regulatory Affairs

Education:

BS Degree and Master of Engineering Degree in Mechanical Engineering, University of Louisville 1979

Experience:

1979 - 1986 U.S. Navy Submarine Officer

CPNPP Experience:

Results Engineer

Engineering Supervisor, Technical Support

Engineering Supervisor, Compliance and Technical Programs

Unit 2 Licensing Manager

Regulatory Compliance Manager

Manager, Nuclear Licensing

Manager, Regulatory Affairs

Donna Christiansen - Director, Nuclear Training

Education:

Georgia Institute of Technology, Atlanta, Georgia – Bach. Nuclear Engineering, 1986 Augusta College, Augusta, Georgia - Masters in Business Administration, 1995

Experience:

04/86 - 10/87	Nuclear Engineer, Nuclear Assurance Corporation
11/87 - 08/90	Reactor Engineer, Plant Vogtle
08/90 - 01/93	Senior Engineer, Operations, Plant Vogtle
01/93 - 06/96	Shift Support Supervisor, Operations, Plant Vogtle
07/96 - 09/97	Nuclear Plant Instructor, Training, Plant Vogtle
09/97 - 04/02	Senior Engineer, Plant Farley
04/02 - 01/04	Shift Support Supervisor, Operations, Plant Farley
01/04 - 10/04	Shift Supervisor, Operations, Plant Farley
10/0 - 05/09	Nuclear Operations Training Supervisor, Plant Farley
05/09 - 03/13	Training Manager, Plant Farley
03/13 - 01/14	Site Readiness Director, Plant Farley
01/14 - 12/15	Maintenance Manager - Contracts, Plant Farley
01/16 -	Director, Nuclear Training, CPNPP

Doyce W. McGaughey - Director, Operations

Education:

June 2013 - Thomas Edison State College Undergraduate Studies - BSAST Nuclear Engineering Technology

Experience:

1976 -	US Navy Mechanical Operator - Nuclear
1985 -	Auxiliary Operator
1992 -	Reactor Operator
1995 -	Unit Supervisor
2001 -	Supervisor, Operations Training
2004 -	Shift Manager
2008 -	Operations Support Manager
2010 -	Operations Performance Improvement Manager
2012 -	Interim Shift Operations Manager
2013 -	Director, Nuclear Performance Improvement
2015 -	Director, Operations

Michael G. Stakes - Director, Maintenance

Education:

BS Electrical Engineering, 1985

Louisiana State University

License/Certifications:

CPNPP Plant Certification 1996

CPNPP SRO License 2005

Experience:

1985 -	Maintenance Engineer
1987 -	Systems Completion Manager for CPNPP Unit 1 Startup
1989 -	Electrical Systems Engineer
1990 -	Predictive Maintenance - Thermographer
1990 -	Maintenance Support Supervisor
1997 -	On-Loan to the Institute of Nuclear Power Operations (INPO) as a maintenance and work management evaluator
1999 -	Operations Manager administrative assistant
2005 -	CPNPP SRO /STA - Fuel Handling Supervisor
2012 -	Shift Manager / STA
2013 -	Shift Operations Manager
2015 -	Director, Maintenance

Steven K. Sewell - Director, Organizational Effectiveness

Education:

Bachelor of Nuclear Engineering, Georgia Institute of Technology - 1986

Experience:

1981 -	System Engineer-coop, Georgia Power Company, Plant E.I. Hatch.
1983 -	Licensing Engineer-coop, Georgia Power Company, Plant E.I. Hatch.
1986 -	Reactor Engineer, CPSES.
1990 -	Licensed as Senior Reactor Operator, CPSES.
1990 -	Operations Unit Supervisor (SRO) STA.
1998 -	Assigned as Operations Shift Manager, CPSES.
2004 -	Assigned as Nuclear Training Manager, CPSES.
2008 -	INPO Team Manager - Plant Evaluations.
2009 -	Assigned as Director, Operations, CPNPP.
2012 -	Assigned as Director, Organizational Effectiveness.
2013 -	Assigned as Plant Manager
2015 -	Director, Organizational Effectiveness

Deborah Farnsworth - Director, Performance Improvement

Education:

Bachelor's Degree in Chemical Engineering, Texas A&M University

Experience:

01/00 - 12/01	Chemistry, Diagnostics, and Materials Engineering Group, Westinghouse Waltz Mill
01/02 - 01/06	Senior Chemist, CPNPP
01/06 - 11/11	Senior Reactor Operator, CPNPP
11/11 - 06/13	Equipment Reliability Coordinator, CPNPP
06/13 - 02/15	Work Control Manager, CPNPP
02/15 -	Director, Performance Improvement, CPNPP

James L. Patton - Manager, Nuclear Oversight

Education:

BS Agricultural Education - Texas A&M University - 1975

Experience:

1976 - Employed by Brown & Root, Inc. as a Quality Control Inspector.

1978 - Assigned as Mechanical Inspection Lead (Supervisor).

1980 - Assigned as Quality Mechanical Superintendent.

1984 - Employed by TU Electric as a Senior QA Technician, CPSES.

1987 - Assigned as Quality Program Supervisor, CPSES.

1990 - Assigned as Senior Nuclear Specialist, CPSES.

2006 - Assigned as Quality Assurance Surveillance Supervisor, CPSES.

2009 - Assigned as Manager Quality Assurance, CPNPP.

Activities:

Senior Member, American Society for Quality - Certified as a Quality Engineer and Quality Auditor

Robert Deppi - Director, Engineering Projects and Support

Education:

Bachelor Degree in Dynamics of Organizational Behavior, Saint David's University

Experience:

1981 - 1986	Reactor Operator, Limerick
1987 - 1991	Senior Reactor Operator, Limerick
1992 - 1993	Work Control Center Supervisor, Limerick
1996 - 1997	Work Control Center Supervisor, Limerick
1997 - 1998	Work Control Director, Clinton
1998 - 2000	COMED Nuclear Generation Group Operation Manager
2000 - 2001	EXELON Byron Station Nuclear Oversight Manager
2002 - 2003	PSEG Site Work Integration Manager
2003 - 2004	FENOC Fleet Operations Programs and Process Manager
2004 - 2006	FENOC Fleet Operation Support, Operations Manager
2006 - 2007	FENOC Fleet Operation Support, Work Control Manager
2008 - 2011	FENOC Manager Major Equipment Reliability Program, Perry Nuclear Power Plant
2011 - 2015	FENOC Fleet Manager, Strategic Industry Initiatives (Fukushima Program)
07/15 -	Director, Engineering Projects and Support, CPNPP

John Taylor - Director, Site Engineering

Education:

Bachelor of Science degree in Mechanical Engineering, Texas A&M University

Master of Science degree in Engineering Technology, University of North Texas

Experience:

08/80 - 08/84	U.S. Navy Submarine Officer
08/84 - 10/86	Staff Engineer, CPNPP
10/86 - 02/89	Supervisor, Mechanical Engineering Services
02/89 - 02/99	Procurement Engineering Supervisor
02/99 - 10/02	Design Bases Engineering Supervisor
10/02 - 04/06	System Engineering Smart Team 2 Manager
04/06 - 08/10	Plant Reliability Manager
07/10 - 06/13	Technical Support Engineering Manager
07/13 - 11/14	Manager System Engineering
11/14	Director, Site Engineering

Deborah O'Connor - Manager, Radiation Protection

Education:

Bachelor of Science in Nuclear Technology, New York State Regents College

CPNPP Experience:

01/93 - 05/96	Radiation Protection Technician
05/96 - 05/99	Senior Health Physicist
05/99 - 10/07	Radiation Protection Supervisor - Dosimetry
10/07 - 01/10	Radiation Protection Supervisor - ALARA
09/14 - 02/15	Nuclear Oversight Rotational Position
02/15	Manager, Radiation Protection

13.2 TRAINING

13.2.1 NUCLEAR TRAINING PROGRAM

CPNPP has a training program which is systems approach to training (SAT) based in accordance with the requirements of 10 CFR parts 50 and 55. CPNPP will comply with the SAT-based training requirements by maintaining accreditation as conferred by the National Nuclear Accrediting Board on 10/25/90. Program descriptions are contained in sections 13.2.1.1 and 13.2.1.2.

13.2.1.1 Training Programs for Licensed Personnel

1. Replacement License Training

   Replacement license training will provide SAT-based training to reactor operator (RO) and senior reactor operator (SRO) candidates in accordance with 10 CFR 55. Replacement license training will comply with the SAT-based training requirements by maintaining accreditation.

2. License Operator Requalification Training

   License operator requalification training will provide SAT-based requalification training to ROs and SROs in accordance with 10 CFR 55. License operator requalification training will comply with the SAT-based training requirements by maintaining accreditation.

3. Simulator Training

   ROs and SROs may perform control manipulations on the simulator required as part of the training in section 13.2.1.1.1 and 13.2.1.1.2. Training will maintain a certified simulator in accordance with the provisions of 10 CFR 55.

4. Training of Licensed Supervisors

   The shift supervisor training program will comply with the SAT-based training requirements, and consists of training in those administrative duty areas above and beyond licensed duties.

5. Training for Shift Technical Advisors

   Shift Technical Advisors not meeting the dual-role SRO/STA requirements as described in Section 13.1.2.1.2, will be trained and qualified in accordance with Option 2 of the Commission Policy Statement on Engineering Expertise (50 FR 43621, October 28, 1985). Shift technical advisor training will comply with the SAT-based training requirements.

13.2.1.2 Training Programs for Nonlicensed Personnel

1. Training for Mechanical Maintenance Personnel

   Mechanical maintenance training will provide SAT-based training to mechanical maintenance personnel in accordance with 10 CFR 50.120. Mechanical maintenance training will comply with the SAT-based training requirements by maintaining accreditation.

2. Training for Electrical Maintenance Personnel

   Electrical maintenance training will provide SAT-based training to electrical maintenance personnel in accordance with 10 CFR 50.120. Electrical maintenance training will comply with the SAT-based training requirements by maintaining accreditation.

3. Training for Instrument and Control Personnel

   Instrument and control training will provide SAT-based training to instrument and control personnel in accordance with 10 CFR 50.120. Instrument and control training will comply with the SAT-based training requirements by maintaining accreditation.

4. Training for Radiation Protection Personnel

   Radiation protection training will provide SAT-based training to radiation protection personnel in accordance with 10 CFR 50.120. Radiation protection training will comply with the SAT-based training requirements by maintaining accreditation.

5. Training for Chemistry Personnel

   Chemistry training will provide SAT-based training to chemistry personnel in accordance with 10 CFR 50.120. Chemistry training will comply with the SAT-based training requirements by maintaining accreditation.

6. Training for Non-licensed Operators

   Non-licensed operator training (plant equipment operators) will provide SAT-based training to non-licensed operators in accordance with 10 CFR 50.120. Non-licensed operator training will comply with the SAT-based training requirements by maintaining accreditation.

7. Training for Engineering Support Personnel

   Engineering support training will provide SAT-based training to engineering support staff in accordance with 10 CFR 50.120. Engineering support training will comply with the SAT-based training requirements by maintaining accreditation.

8. Training on the Emergency Diesel Generator

   Operations and maintenance personnel responsible for emergency diesel generator performance will receive training on the EDG commensurate with the level of expertise required.

13.2.1.3 General Employee Training

1. All employees (and others) who require unescorted access to the Protected Area of the station will receive training in the following areas:

   a. General description of plant and facilities

   b. Emergency Plan and procedures

   c. Fire Protection Program and Procedures

   d. Security Requirements and Practices

   e. Safety Program

   f. Quality Assurance Program

   g. Radiological Protection Program

2. All employees (and others) who have unescorted access to Radiation Controlled Areas of the station will receive in-depth instruction in all aspects of radiation protection and, as required, respiratory protection. Subject material will include but will not be limited to the following:

   a. Handling radioactive material

   b. Controls and access requirements

   c. Biological effects of ionizing radiation

3. Some employees receive, as part of their specialty training, a General Plant Information Course. This lecture course consists of reactor theory, primary plant systems, secondary systems, and instrumentation and control.

4. Review of appropriate department and station procedures are specified in department specialty training programs.

13.2.1.4 Fire Protection Training

The CPNPP Fire Protection Training Program is described in section 13.3B.

13.2.2 REPLACEMENT AND RETRAINING

13.2.2.1 Licensed Operators - Requalification Training

NOTE: All replacement and retraining requirements are embedded within the accredited programs as described in sections 13.2.1.1 and 13.2.1.2.

13.2.3 APPLICABLE NRC DOCUMENTS

The NRC Regulations, Regulatory Guides and Reports listed below were used to provide guidance in the area of training for staff personnel. Compliance to these items is indicated below.

1. 10 CFR Part 50, “Licensing of Production and Utilization Facilities.” Full compliance in the area of training.

2. 10 CFR Part 55, “Operators’ Licenses.” Full compliance in the area of training.

3. 10 CFR Part 19, “Notices, Instructions, and Reports to Workers; Inspection.” Full compliance in the area of training.

4. Regulatory Guide 1.8, “Personnel Selection and Training.” Compliance in this area is discussed in Section 12.5.1, “Organization,” and 13.1.3.1, “Minimum Qualification Requirements, and 13.2, “Training”.”

5. Regulatory Guide 8.2, “Guide for Administrative Practices in Radiation Monitoring.” Full compliance in the area of training.

6. Regulatory Guide 8.8, “Information Relevant to Maintaining Occupational Radiation Exposure As Low As Reasonably Achievable (Nuclear Power Reactors).” Full compliance in the area of training except as noted in Item #4 above.

7. Regulatory Guide 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low As Is Reasonable Achievable.” Full compliance in the area of training.

8. Regulatory Guide 8.13, “Instruction Concerning Prenatal Radiation Exposure”. Full compliance in the area of training.

9. Generic Letter 87-07, “Information Transmittal of Final Rulemaking for Revisions to Operator Licensing - 10 CFR 55 and Conforming Amendments”. Used as guidance only.

10. NUREG-0737,” Clarification of TMI Action Plan Requirements”, Published November, 1980.

11. 50 FR 43621, “NRC Policy Statement on Engineering Expertise on Shift”. Full compliance with Option 2.

13.3 EMERGENCY PLANNING

Emergency planning for Comanche Peak Nulclear Power Plant is directed toward mitigating the consequences of emergencies and provides reasonable assurance that appropriate measures can and will be taken to protect health and safety and prevent damage to property in the event of an emergency. It is done with the paramount objective to protect the health and safety of the general public, persons temporarily visiting or assigned to the station, and station employees.

The CPNPP Emergency Plan is described in Section 13.3A below.

The CPNPP Fire Protection Program is described in Section 13.3B below.

13.3A EMERGENCY PLAN

The CPNPP Emergency Plan has been upgraded in order to assure that adequate protective measures can and will be taken in the event of a radiological emergency. The guidance of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," was used to extensively revise the Emergency Plan which is provided in a separate binder titled "Comanche Peak Nulclear Power Plant Emergency Plan."

13.3B CPNPP FIRE PROTECTION PROGRAM

The senior management position responsible for this program is the Site Vice President. The Site Vice President has delegated to the Operations Review Committee the responsibility to assess the effectiveness of the fire protection program which is accomplished through periodic audits as discussed in FSAR Section 17.2.1.3.2. Recommendations and the findings from these audits are reported to the Site Vice President.

The CPNPP Fire Protection Program meets the requirements of the D. B. Vassallo letter of August 1977 entitled "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," as described in the following sections.

The CPNPP Fire Protection Program is established to ensure that a fire will not prevent safe plant shutdown and will not endanger the health and safety of the public. Fire protection at CPNPP is accomplished by using a defense-in-depth approach to include fire detection and extinguishing systems and equipment, (See Section 9.5.1) administrative controls and procedures, and trained personnel. Procedures for implementing the CPNPP Fire Protection Program will be developed prior to Start-Up. CPNPP is committed to meeting the requirements of the Fire Protection Report.

13.3B.1 ORGANIZATION

13.3B.1.1 Plant Manager (See Section 13.1.1.2.1)

The Plant Manager has the responsibility for maintenance, and implementation of the Fire Protection Program for operations at CPNPP. He has delegated this responsibility to the Director, Maintenance.

13.3B.1.2 Director, Maintenance

The Director, Maintenance is responsible to the Plant Manager (see Section 13.1.1.2.1) for the maintenance and implementation of the Fire Protection Program. The Director, Maintenance has assigned to the Maintenance Team Manager the responsibility of assuring the overall maintenance and implementation of the Fire Protection Program.

13.3B.1.3 Director, Site Engineering

The Director, Site Engineering has the responsibility for the development of the Fire Protection Program at CPNPP. The Director, Site Engineering has assigned to the Manager, Technical Support the overall responsibility for the development of the CPNPP Fire Protection Program.

13.3B.1.4 Manager, Technical Support

The Manager, Technical Support is responsible to the Director, Maintenance for the overall development and coordination of the CPNPP Fire Protection Program. Duties and responsibilities include:

1. The continued maintenance of the Fire Protection Licensing Documents and evaluation of regulatory requirements for impact.

2. Review and approve Fire Protection Design documents, licensing documents, and Station Procedures including revisions, to verify technical accuracy and regulatory compliance.

3. The continued implementation of the Fire Protection Engineering requirements such that modifications and changes do not impact the ability to safely shutdown the plant in the event of a fire.

The Manager, System Engineering is responsible to the Director, Site Engineering for technical support of the Fire Protection Program in relation to the review and implementation of design modifications and the resolution of system issues.

13.3B.1.5 Maintenance Team Manager

The assigned Maintenance Team Manager is responsible to the Director, Maintenance for the overall implementation of the Fire Protection Program. Duties and responsibilities include:

1. Assures the implementation of periodic inspections to minimize the amount of combustibles in safety-related areas; determine the effectiveness of housekeeping practices; assure the availability and acceptable condition of all fire protection systems/ equipment, and assures that prompt actions are taken to correct conditions adverse to fire protection and preclude their recurrence.

2. Ensures that periodic testing and maintenance of fire protection systems and equipment is being performed and evaluated for availability and acceptability.

3. The development of the Fire Protection Program administrative procedures and the fire protection systems and equipment testing and maintenance requirements.

4. Assures the implementation of the administrative procedures of the Fire Protection Program such that the ability to safely shutdown the plant in the event of a fire is not compromised due to hot work, systems or equipment being impaired, compensatory measures or the control of transient materials and/or flammable/combustible liquids and gases.

5. To assist the Nuclear Training Manager in the development and implementation of fire protection training programs for personnel and the fire brigade at CPNPP.

13.3B.1.6 Director, Nuclear Training

The Director, Nuclear Training reports to the Site Vice President (see Section 13.1.1.2.1) and assists in the development and implementation of fire protection training programs for operating personnel and the fire brigade at CPNPP as requested. The Director, Nuclear Training documents and maintains records of the fire protection training of operations personnel and fire brigade.

13.3B.2 CPNPP FIRE BRIGADE

The Comanche Peak Nulclear Power Plant will have an organized fire brigade to deal with fires and related emergencies when they occur. The size and nature of the CPNPP fire brigade will be based upon an evaluation of the potential magnitude of a fire emergency at CPNPP and the availability of fire fighting assistance from offsite departments. However, CPNPP will be self-sufficient with respect to fire fighting activities to protect critical areas within the plant.

The CPNPP fire brigade organization will be based upon a team concept. The brigade will be organized into fire teams consisting of five (5) men each. This fire team size is consistent with the equipment that must be put into service during a fire emergency (2 1/2" hose station, 1 1/2" hose station, wheeled and hand held portable extinguishers). Each fire team will have a designated fire team leader to direct the action of the fire team. The fire team leader will also maintain close communication with the Shift Manager, keeping him apprised of the situation at the fire site. Two (2) men will perform the primary fire fighting function, i.e. operate the fire suppression equipment. The largest equipment to be used by these two men is the 2 1/2 inch fire hose located on the Turbine Building Operating Deck. The remaining two (2) men will be utilized for:

1. operating additional fire suppression equipment,

2. operating portable smoke ejectors, or

3. resupplying fresh bottles for the breathing apparatus.

A sufficient number of operations personnel will receive fire brigade training and will qualify to be members of the CPNPP fire brigade. A five (5) man fire team consisting of the fire team leader and four (4) additional personnel will be on duty each working shift and at periods when the plant is shutdown. The Fire Brigade shall not include the Shift Manager and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency. Other plant personnel will also receive fire brigade training and likewise be qualified members of the CPNPP fire brigade.

The Fire Brigade may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty crew members provided immediate action is taken to restore the brigade composition to within the minimum requirements. This provision does not permit the brigade to be unmanned below the minimum upon shift change due to an oncoming member being late or absent.

The on-duty Shift Manager will assume overall responsibility in the event of a fire emergency. He will be responsible for the following actions based upon his assessment of the magnitude of the fire emergency from reports received from the fire team leader:

1. Safe shutdown of the plant if required.

2. Implementation of the Emergency Plan.

3. Notification of management.

4. Requesting assistance from off duty fire teams if deemed necessary.

5. Contacting local fire departments if required.

If the decision is made to implement the emergency plan as a result of a fire emergency, the Shift Manager is designated the Emergency Coordinator until relieved by a designated alternate. However, the Shift Manager will continue to receive reports from the team leader at the fire site while coordinating other emergency activities with the Emergency Coordinator.

To qualify as a member of the CPNPP fire brigade, individuals must be available to answer alarms and to attend required training sessions.

13.3B.2.1 Fire Brigade Training

A training program will be established to assure that the capability to fight potential fires is developed and documented. The program will consist of a classroom instruction program supplemented with periodic classroom retraining and practice in fire fighting and fire drills. Classroom instruction and training will be conducted by qualified individuals knowledgeable of fighting the types of fires that could occur in the plant and in using available fire fighting equipment.

13.3B.2.1.1 Classroom Instruction

Fire brigade members will receive classroom instruction in fire protection and fire fighting techniques at planned meetings. Instruction will include:

1. Identification of fire hazards and associated types of fires that could occur in the plant and an identification of the location of such hazards.

2. Identification of the location of fire fighting equipment for each fire area, and familiarization with layout of the plant including ingress and egress routes to each area.

3. The proper use of available fire fighting equipment and the correct method of fighting each type of fire, to include electrical fires, cable and cable tray fires, hydrogen fires,

flammable liquids, waste/debris fires, fires involving radioactive materials, and record file fires.

4. Review of the CPNPP fire fighting plan with coverage of each individual's responsibilities.

5. The proper use of communication, lighting, ventilation and emergency breathing equipment.

6. The direction and coordination of the fire fighting activities.

7. The toxic and radiological characteristics of expected products of combustions.

8. The proper method of fighting fires inside buildings and tunnels.

9. Review of fire fighting procedures and procedure changes.

10. Review of fire protection-related plant modifications and changes in fire fighting plans.

Instruction will be provided for all employees once a year and repeated on an annual basis. The instruction will include, as appropriate, the fire protection plan, evacuation routes and procedure for reporting a fire.

Instruction will be provided for security personnel including entry procedures for outside fire departments and crowd control for people exiting the station.

Instruction will be provided for construction personnel and temporary employees to include alarm responses, evacuation routes and procedure for reporting fires.

Off-site fire organizations will receive training on basic radiation principles and practices and typical radiation hazards that may be encountered when fighting fires.

13.3B.2.1.2 Practice

Practice sessions will be held on an annual basis for fire brigade members and will include the proper method of fighting various types of fires similar to those which might occur in a nuclear power plant. These sessions will provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus.

Emphasis will be placed on training brigade members to fight fires in safety-related areas.

13.3B.2.1.3 Drills

Fire brigade drills will be conducted on a quarterly (every 3 months) basis at CPNPP. Drills will be of two types: announced and unannounced. Drills will allow fire team individuals to practice together as a team. A sufficient number of drills will be conducted in a period of three months so that all fire teams will participate in at least one drill. Each individual member of a fire team will participate in at least two drills per year. Offsite fire organizations will be included in a fire brigade drill at least annually. Training objectives will be established prior to the drill. Afterwards, to determine how well the training objectives have been met, the drill will be critiqued to include:

1. Assessment of fire alarm effectiveness, time required to notify and assemble fire brigade, and selection, placement and use of equipment.

2. Assessment of brigade leader's effectiveness in directing the fire fighting effort.

3. Assess brigade member's knowledge of fire fighting strategy, procedures, and use of equipment in the area assumed to contain the fire.

Employees who receive plant access training are instructed in fire response including evacuation. Evacuation of employees (site evacuation / accountability) is also exercised as an element of the CPNPP Emergency Plan.

13.3B.2.1.4 Records

Records of training provided for each fire brigade member will be maintained to assure that each member of the fire brigade receives training in all parts of the program.

13.3B.3 FIRE FIGHTING PROCEDURES

Fire fighting procedures will be established to cover such items as notification of a fire, fire emergency procedures, and coordination of fire fighting activities with local fire departments. The procedures will identify:

1. Actions to be taken by the individual discovering the fire, such as notification of Control Room, attempt to extinguish fire, and activation of local fire suppression systems.

2. Actions to be taken by Control Room personnel, such as sounding fire alarms and notifying the Shift Manager/fire brigade leader of type, size and location of fire.

3. Actions to be taken by fire brigade after notification of a fire, including location to assemble, directions given by fire brigade leader, and responsibilities of brigade members such as selection of fire fighting and protective equipment and use of pre-planned strategies for fighting fires in specific areas.

4. Actions to be taken by plant management and Security after notification of fire.

5. Actions to be taken that will coordinate fire fighting activities with offsite fire departments including identification of person responsible for assessing situation and calling in local fire department's assistance if deemed necessary.

6. The strategies established for fighting fires in safety-related areas and areas presenting a hazard to safety-related equipment; strategies such as identification of combustibles in each plant zone covered by a fire fighting procedure type fire extinguishers best suited for controlling the fires with the combustible loadings of the zone, and instructions for plant operators and general plant personnel during fire.

13.3B.4 ADMINISTRATIVE PROCEDURES AND CONTROLS

Administrative procedures and controls will be established to ensure the reliable performance of fire protection personnel, systems and equipment. Effective measures will be established to control the use and storage of combustibles and ignition sources.

13.3B.4.1 Control of Combustibles

Effective administrative controls will be established to minimize the amount of combustibles that safety-related areas may be exposed to during operation or maintenance periods. These controls will govern the following:

1. Proper storage and handling of flammable gases and liquids, HEPA and charcoal filters, dry, unused ion exchange resins or other combustible supplies in safety-related areas.

2. Transient fire loads during maintenance and modifications such as combustibles and flammable liquids, wood and plastic materials in buildings containing safety-related systems or equipment. This control will require an in-plant review of work activities to identify transient fire loads. The supervisor or foreman responsible for reviewing the work activity will specify any required additional fire protection.

3. Waste, debris, scrap, and oil spills resulting from a work activity in a safety-related area will be minimized while work is in progress and removed upon completion of the activity.

4. Periodic inspection for accumulation of combustibles.

13.3B.4.2 Control of Ignition Sources

Effective administrative controls will be established to protect safety-related equipment from fire damage or loss resulting from work involving ignition sources, such as welding, cutting, grinding or open flame work. Administrative controls will prohibit the use of open flame or combustion smoke for leak testing and other ignition sources in certain areas.

Administrative controls will be established to ensure that the following precautions are taken:

1. Welding, cutting, grinding or open flame work will be authorized by the responsible foreman or supervisor through a work permit. The responsible foreman or supervisor will have received sufficient fire fighting and fire prevention training covering anticipated fires, such as electrical fires, fires in cables and cable trays, hydrogen fires, hydrocarbon fires, solvent fires, waste/debris fires, and record file fires.

2. Before work is performed on or near safety-related equipment, combustible materials located in the vicinity of cutting, welding or open flame work will be removed or protected by curtains, metal guards, or flameproof curtains. Fire extinguishers or other fire fighting equipment will be provided in the vicinity of the work location, if needed.

3. Smoking will not be allowed in safety-related areas. Smoking is only allowed in outside areas as outlined by the Company Smoking Policy.

13.3B.5 QUALITY ASSURANCE PROGRAM

The CPNPP QA Program establishes the quality assurance requirements and controls to be implemented throughout the testing and operation phases of CPNPP. This program defines responsibility and authority and prescribes measures for the control and accomplishment of activities affecting the quality and operation of safety-related structures, systems and components. The structures, systems, and components covered by the Operations Quality Assurance Program are listed in Appendix 17A. These include both nuclear safety-related items subject to 10CFR50, Appendix B QA and non safety-related items which are not subject to Appendix B (e.g. Fire Protection). The provisions of the plan apply to all activities such as operating, maintaining, repairing and modifying which could affect the safety-related functions of safety-related systems and components. The preparation, review and approval of Fire Protection Program procedures will be handled in the same manner as safety-related procedures.

The quality assurance to be applied to the CPNPP Operation's Fire Protection Program is as follows:

1. Design Control and Procurement Document Control.

   Measures are established to assure that all design-related guidelines of the Branch Technical Position are included in design and procurement documents and that any deviations therefrom are controlled. These procedures include provisions, as necessary, to ensure that:

   a. Design and procurement document changes are subject to the same controls that were applicable to the original design.

   b. Design documents specify quality requirements or reference quality standards, as necessary, such as appropriate fire protection codes.

   c. Design and procurement documents are reviewed by qualified personnel to assure inclusion of appropriate fire protection requirements.

2. Instructions, Procedures and Drawings.

   Inspections, tests, administrative controls, fire drills and training, that govern the fire protection program is prescribed by documented instructions, procedures or drawings and are accomplished in accordance with the applicable document.

3. Control of Purchased Material, Equipment and Services.

   Measures will be established to assure that purchased material, equipment and services conform to the procurement documents.

4. Inspection.

   A program for independent inspection of activities affecting fire protection is established and executed by, or for, the organization performing the activity to verify conformance with documented installation drawings and test procedures for accomplishing the activity.

5. Test and Test Control.

   A test program is established to implement the operational surveillance testing required by the CPNPP technical specifications on fire protection. These tests are performed by written and approved procedures and will be subject to the same controls required for surveillance tests for safety related equipment. Test results will be properly evaluated and acted on.

6. Inspection, Test, and Operating Status.

   Measures will be established to provide for the identification of items that have satisfactorily passed required tests and inspections.

7. Nonconforming Items.

   Measures are established to control items that do not conform to specified requirements to prevent inadvertent use or installation.

8. Corrective Action.

   Measures are established to ensure that conditions adverse to fire protection such as failures, malfunctions, deficiencies or deviations, defective components, uncontrolled combustible material and non-conformances are promptly identified, reported, and corrected.

9. Records.

   Records will be prepared and maintained to assure conformance to fire protection requirements. These records will include results of inspections, tests, reviews, and evaluations; nonconformance and corrective action reports; and maintenance and modification records. Record retention requirements will be established.

10. Audits.

    Audits will be conducted and documented to verify compliance with the fire protection program, including design and procurement documents; instructions; procedures and drawings; and inspection and test activities. These audits will be performed by personnel not having direct responsibility for the Fire Protection Program.

13.4 REVIEW AND AUDIT

A program for review and audit of activities affecting station safety during the operational phase has been established. The program provides a system to insure that these activities are performed in accordance with company policy and rules, approved procedures and license provisions. This program will provide review of safety-related plant changes, tests, and procedures.

13.4.1 ONSITE REVIEW

See Section 17.2.1.1.2.1 for Station Operations Review Committee (SORC).

13.4.2 INDEPENDENT REVIEW

Activities affecting station safety occurring during the CPNPP operational phase will be independently reviewed by Nuclear Oversight and/or the Operations Review Committee (ORC).

13.4.2.1 Operations Review Committee (ORC)

See Section 17.2.1.3 for details.

The purpose for ORC review of violations, deviations, and reportable events which require reporting to the NRC in writing is to verify the adequacy of any investigations of such events and the corrective actions taken to prevent or reduce the probability of recurrence of such events.

ORC will also conduct independent reviews of the following:

1. All inspection reports conducted by the NRC and selected reports of audits conducted in accordance with Section 17.2.18.2 to verify compliance with the CPNPP Quality Assurance Manual and CPNPP license requirement.

2. Selected reports and documentation, as specified by the ORC, which pertain to reviews of safety-related activities conducted by the CPNPP Operating Staff. The purpose is to verify the adequacy of the on-site review process and provide an overview of other SORC activities.

3. Other safety-related matters deemed appropriate by ORC members or referred to the ORC by the Station Operations Review Committee.

13.4.3 AUDITS

Audits will be conducted in accordance with Chapter 17 of the FSAR.

13.5 PLANT PROCEDURES AND INSTRUCTIONS

The CPNPP staff is responsible for assuring the safe and efficient operation of the station, under the overall responsibility and direction of the Plant Manager (see Section 13.1.1.2.1). All activities which affect safety-related structures, systems and components will be conducted by detailed, written and approved procedures and instructions. This section identifies the activities that must be conducted by procedures and instructions and provides an appropriate method to develop and approve these procedures and instructions.

13.5.1 ADMINISTRATIVE PROCEDURES

The Plant Manager (see Section 13.1.1.2.1) will develop and implement written administrative procedures that assign the responsibilities and authorities of the CPNPP staff and provide the control measures for the preparation, review, approval, revision and use of all station procedures and instructions which govern quality-related activities. Station procedures and instructions include plant operating procedures, test procedures, equipment control procedures, calibration procedures and instructions, maintenance or modification procedures and instructions, refueling procedures, procurement procedures, analysis procedures, material control procedures and special orders. Administrative procedures will ensure that station procedures and instructions are reviewed by qualified personnel, approved by authorized personnel and distributed to and used by the personnel performing the prescribed activity.

13.5.1.1 Conformance with Regulatory Guide 1.33

The administrative controls utilized during the operations phase, which are described in this section, will be consistent with the provisions of Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation),” Revision 2, February 1978.

13.5.1.2 Preparation of Procedures and Instructions

Preparation of plant operating procedures will take place in approximately the same time frame as the preparation of preoperational and initial startup test procedures. Administrative procedures which govern the assignment of responsibilities for preparation, review and approval of other station procedures and instructions will be prepared initially. Other administrative procedures will be prepared as necessary to implement the operational phase of programs such as security and visitor control, housekeeping, and document control and records management, etc. Operation department administrative procedures and operating procedures necessary for operator training and preparation for operator license examinations will be completed six months prior to fuel loading. All other procedures and instructions will be prepared and approved prior to their use for performing the prescribed safety-related activity.

The station management position designated responsible for a given activity, as prescribed in the Quality Assurance Manual, is also responsible for the preparation of procedures and instructions for that activity. The actual preparation of procedures and instructions may be performed by other Luminant Power personnel or by outside contractors, but the final responsibility lies with the designated responsible position.

The Plant Manager (see Section 13.1.1.2.1) shall approve Station Administrative Procedures. Security Plan implementing procedures and Emergency Plan implementing procedures are approved in accordance with provisions of the CPNPP Security Plans and Emergency Plan,

respectively. All procedures shall be reviewed by qualified personnel and these reviews will be documented. Quality-related procedures and instructions shall be reviewed by at least one individual other than the preparer and approved by an appropriate manager. This designation of the appropriate manager shall be stated in writing by the SORC and approved by the Plant Manager (see Section 13.1.1.2.1).

Changes to approved quality-related procedures and instructions which clearly do not change the intent of the procedure and which require urgent implementation may be approved by two members of the Nuclear Operations staff, at least one of whom is SRO licensed. These changes shall be approved by the original approval authority within 14 days of implementation.

Other changes to procedures and instructions shall be reviewed and approved in the same manner as a permanent revision to that document or as otherwise directed by SORC in writing.

13.5.1.3 Procedures

The Plant Manager (see Section 13.1.1.2.1) will develop and implement station administrative procedures that provide a clear understanding of operating philosophy and management policies. As stated in 13.5.1.2, administrative procedures will be implemented that provide methods for the preparation, review and approval of all other station procedures including permanent procedures, temporary procedures or any procedures that might be of a transient or self-cancelling nature.

Administrative procedures will be developed by the operations department that will provide operations Shift Managers and shift crews with a clear understanding of how they are to conduct plant operations. Included will be procedures that specifically describe who may manipulate the controls of the reactor and who may operate any apparatus or mechanism that might affect the reactivity of the reactor. Procedures also will be prepared that describe the responsibilities of the senior reactor operators who will direct the licensed activities of licensed operators and who will be present at the facility or on call.

An operator requalification program will be implemented as described in Section 13.2.2.1 of the FSAR.

Examples of Operations Department Administration Procedures that may be included are listed in Table 13.5-9.

The operations department will also implement procedures and instructions as required specifying shift manning requirements which will be in accordance with the Technical Specifications. The responsibilities and authorities of the senior licensed personnel shall be delineated such as:

1. Determine the circumstances, analyze the cause, and determine that operations can proceed safely before the reactor is returned to power after a trip or an unscheduled or unexplained power reduction.

2. Accept and respond conservatively to instrument indications unless they are proven incorrect.

3. Adhere to the plant's Technical Specifications.

4. Review routine operating data to assure safe operation.

Procedures and instructions shall prescribe the conduct of shift operational activities including the following:

1. Definition of the specific area where the reactor operator who is at the controls of the unit must remain. This area is shown on Figure 13.5-1 and defines the area of applicability of position C.1.n of Regulatory Guide 1.29 (See Appendix 1A(B)).

2. Measures to control access to the Control Room.

3. Procedures for proper shift relief and turnover.

4. Procedures and instructions for the control of log and record keeping.

During station operation, the Shift Manager shall be responsible for ensuring that equipment control procedures are followed and properly implemented. These procedures will provide control of equipment, as necessary, to maintain personnel safety and reactor safety, and to avoid unauthorized operation of equipment. To secure and identify equipment in a controlled status, measures such as temporary bypass lines, electrical jumpers, lifted electrical leads, and temporary trip point settings shall be controlled by approved procedures which shall include requirements for independent verification where appropriate. A log will be maintained of the current status of temporary modifications which are not specified and controlled by approved procedures or instructions.

The plant operations and technical support departments shall be responsible for developing and implementing procedures, instructions and schedules to describe and control a surveillance inspection program for those areas for which they are responsible. A master surveillance schedule will be established which reflects the status of all surveillance test and inspections required by the Technical Specifications.

The nuclear operations departments will develop and implement administrative procedures that describe and control a preventive maintenance program. These administrative procedures will be written before initial station startup and will provide the general rules for the development of procedures and instructions under the preventive maintenance program. This program will provide for advance planning and scheduling of required routine preventive repair and maintenance activities.

The plant operations and plant engineering departments will also establish administrative procedures and instructions to control and document major repair and modification. Repairs or modifications which may affect the functioning of safety-related structures, systems or components will be performed in a manner to ensure quality at least equivalent to that specified by the original design specifications, materials specifications and inspection requirements.

Operations shall consider potential flooding when authorizing maintenance on systems which utilize circulating water and which could cause flooding.

Administrative procedures and controls will be established to ensure the reliable performance of fire protection personnel, systems and equipment. FSAR Section 13.3, Appendix B, Section 5.0 describes these procedures in more detail.

13.5.2 STATION OPERATING AND MAINTENANCE PROCEDURES

13.5.2.1 Station Operating Procedures

Operating procedures for all anticipated conditions affecting reactor safety will be written prior to initial fuel loading. The format of the station operating procedures ensures the contents include the applicable elements of ANSI N18.7-1976. These procedures will be grouped into the following classifications:

1. Integrated Plant Operating Procedures

2. System Operating Procedures

3. Alarm Procedures

4. Abnormal Conditions Procedures

5. Emergency Operating Procedures [addressed by TMI Action item I.C. 1.]

6. Radwaste Systems Procedures

13.5.2.1.1 Integrated Plant Operating Procedures

These procedures provide guidance for integrated plant operations and they will include major steps to be performed, references to system operating procedures, valve alignments and verification of instrument calibrations and approvals. Check lists will be included where appropriate.

Examples of procedures that may be included in this category are given in Table 13.5-1.

13.5.2.1.2 System Operating Procedures

These procedures include instructions for energizing, filling, venting, draining, startup, shutdown, changing operational modes and other appropriate instructions for the normal operation of safety-related systems. Valve lineups and check-off sheets will be included where necessary to ensure proper system operation.

Examples of procedures that may be included in this category are listed in Table 13.5-2.

13.5.2.1.3 Alarm Procedures

Each main control board annunciator for safety-related parameters will have a written procedure to identify the proper action to be taken by the operator in response to an alarm. Each of these procedures will include the annunciator identification, alarm setpoints and the proper corrective action to be taken.

13.5.2.1.4 Abnormal Conditions Procedures

These procedures will address abnormal operating conditions of safety- related plant systems. Each procedure will contain corrective action instructions for the malfunction involved.

Examples of procedures that may be included in this category are given in Table 13.5-3.

13.5.2.1.5 Emergency Operating Procedures

The Emergency Operating Procedures will be provided to guide operations in order to prevent or lessen the consequences of emergency conditions. These procedures will include automatic actions that will occur in the event of an emergency, immediate operator actions required to prevent or mitigate the consequences of an emergency and subsequent operator actions necessary to stabilize plant conditions. Immediate operator actions required to be memorized will be identified.

The Emergency Operating Procedures will be written to provide for a conservative course of action on the part of the operator and will be sufficiently flexible to accommodate variations.

Examples of procedures that may be included in this category are listed in Table 13.5-4.

13.5.2.1.6 Radwaste Systems Procedures

These procedures will be instructions for the operation of radioactive liquid, solid and gaseous waste systems and will be written to provide guidance for collection, storage, processing and discharge of these materials. Sound radioactive waste management will be incorporated into these procedures to support the effort to minimize radiation exposure and precisely control the release of radioactive material to the environment.

Examples of procedures that may be included in this category are given in Table 13.5-5.

13.5.2.2 Other Procedures

13.5.2.2.1 Maintenance and Modification Procedures

Maintenance or modification that may affect the functioning of safety- related structures, systems, or components shall be performed in accordance with applicable codes, bases, standards, design requirements, material specifications, and inspection requirements.

Maintenance of safety-related equipment will be properly pre-planned and performed in accordance with written procedures, written instructions, or drawings appropriate to the circumstances (for example, skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineation in a written procedure). It is the responsibility of the Director, Maintenance to implement a maintenance program for safety-related mechanical and electrical equipment and instruments and controls. It is the responsibility of the Director, Site Engineering to assist in performance testing of safety-related mechanical and electrical equipment to assure adequate levels of performance.

General rules for the control and administration of the maintenance program will be written before fuel loading. These general rules will form the basis for developing the repair or replacement procedures and instructions at the time of failure.

Procedures and instructions will be written early in plant life for maintenance of safety-related equipment expected to require recurring maintenance. The causes of malfunctions shall be determined, evaluated and recorded. The type and depth of the evaluation shall be commensurate upon the significance of the event or malfunction and shall be performed per procedural and program requirements. Since the probability of failure is usually unknown and the time and mode of failure are usually unpredictable, procedures and instructions will not generally be written for repair of most equipment prior to failure. As experience is gained in operation of the plant, routine maintenance will be altered to improve equipment performance and repair procedures and instructions will be written and improved as required.

A preventive maintenance schedule will be developed which will describe the frequency and type of maintenance to be performed. A preliminary schedule will be developed early in plant life and will be refined and changed as experience with the equipment is gained.

Maintenance will be scheduled so as not to jeopardize the safety of the reactor. Scheduling shall consider the possible safety consequences of concurrent or sequential maintenance, testing, or operating activities. Equipment required to be operable for the mode in which the reactor exists will be available, and maintenance will be performed in a manner such that the license limits are not violated.

A managed maintenance program was developed to provide the plant staff with the maintenance data and information systems necessary to support proper planning and management of the maintenance activities. This is accomplished by a systematic evaluation of each plant component in which all maintenance activities are assessed. Examples of these resources are: manpower, radiation exposure, special tools, spare tools, spare parts, procedure number, and plant condition required for performing the activity.

Once all maintenance activities have been identified, two sets of maintenance plans are generated. The first set is the on-line preventive maintenance plan which includes all of those maintenance activities which can be performed with the plant at power. These activities are scheduled on a computer with various printouts and worksheets for the craft and supervisory personnel.

The second set of plans includes the outage-related work which will normally be performed concurrent with refueling. These activities, along with the refueling sequence are plotted on a computer which is used for managing the outage.

A significant point to note is that, because of inservice inspection requirements, the outage plan is repetitive with a ten-year cycle. The plant staff has completed the outage plans for the first ten-year cycle and, because of the repetitive nature of the work, has a plan for each year of commercial operation throughout the life of the plant. The program is designed to be an active program which will be updated as plant conditions and requirements change.

Off-the-shelf components shall be used only when the proper quality assurance documents are available or when the required quality assurance can be obtained by inspection and testing prior to being placed in service. Major modifications to safety-related equipment shall be designed

and performed in accordance with applicable codes, standards, bases, design requirements, materials specifications and inspection requirements.

13.5.2.2.2 Plant Radiation Protection Procedures and Instructions

Detailed written and approved procedures and instructions will be used by the Comanche Peak Station Nuclear Operations Organization to ensure that occupational radiation exposure is maintained as low as reasonably achievable. It is the responsibility of the Radiation Protection Manager to prepare and maintain the plant radiation protection procedures and instructions. Careful administrative control of the use of these procedures and instructions will ensure that a sound health physics philosophy becomes an integral part of station operation and maintenance. Plant radiation protection procedures and instructions will be developed for activities such as those listed in Table 13.5-6.

13.5.2.2.3 Emergency Preparedness Procedures

The Manager, Nuclear Emergency Planning is responsible for preparing and maintaining procedures that implement the protective measures outlined in the CPNPP Emergency Plan. A list of implementing procedures is maintained in the CPNPP Emergency Plan.

13.5.2.2.4 Chemical and Radiochemical Control Procedures and Instructions

The preparation of detailed, written and approved chemical and radio- chemical procedures and instructions are the responsibility of the Manager, Nuclear Chemistry.

These procedures and instructions will ensure primary and secondary side chemical/ radio-chemical quality, protection of component integrity, and promotion of efficient plant operation. Examples of areas to be covered by written and approved procedures and instructions are listed in Table 13.5-8.

13.5.2.2.5 Instrument Calibration and Test Procedures and Instructions

The Director, Maintenance is responsible for preparing procedures and instructions for proper control and periodic calibration of plant measuring and test equipment to maintain accuracy within necessary limits and to confirm adequacy of calibration frequency.

Specific procedures are prepared for surveillance tests performed on safety-related equipment and instrumentation. These procedures have provisions for assuring measurement accuracies are adequate to keep safety parameters within operational and safety limits. A master surveillance schedule reflecting the status of all planned in-plant surveillance testing is maintained. Control measures exist to assure appropriate documentation, reporting, and evaluation of test results.

13.5.2.2.6 Material Control Procedures and Instructions

Procedures and instructions will be provided for the proper procurement, documentation, and control of safety-related materials and components necessary for plant maintenance and modification. The procedures will be sufficiently detailed to ensure that purchased materials and components associated with safety-related structures or systems are:

1. Purchased to specifications and codes which ensure performance at least equivalent to the original equipment;

2. Produced or fabricated under quality control which ensures performance at least equivalent to that of the original equipment;

3. Properly documented to show compliance with applicable specifications, codes and standards;

4. Properly inspected, identified, and stored to provide protection against damage or misuse;

5. Properly controlled to ensure the identifications, segregation, and disposal of non-conforming material.

13.5.2.2.7 Security Procedures

It is the responsibility of the Manager, Nuclear Security to prepare and maintain detailed, written and approved procedures to implement the security plan. These procedures will supplement the physical barriers and other features designed to control access to the station and, as appropriate, to vital areas within the station. Information concerning specific design features and administrative provisions of the security plan is accorded limited distribution on a need-to-know basis.

13.5.2.2.8 Fuel Handling Procedures

The Director, Engineering Support and Director, Operations are responsible for developing fuel handling procedures to include receipt and receipt inspection, fueling/refueling and fuel handling, storage of fuel, and fuel shipment.

13.5.2.2.9 Fire Protection Procedures

The Manager, Technical Support is responsible for the development of procedures for fire protection. These procedures are described in more detail in FSAR Section 13.3B.

13.5.2.2.10 Test Procedures

The Manager, Startup was responsible for the preparation and implementation of procedures for prerequisite and preoperational acceptance testing. The Director, Site Engineering is responsible for design modification acceptance, station performance, initial startup tests, and those surveillance tests assigned to the System Engineering Department.

TABLE 13.5-1
INTEGRATED PLANT OPERATING PROCEDURES

- Plant Heatup From Cold Shutdown to Hot Standby
- Plant Startup From Hot Standby to Minimum Load
- Power Operations
- Plant Shutdown From Minimum Load to Hot Standby
- Plant Cooldown From Hot Standby to Cold Shutdown
- Maintaining Hot Standby
- Plant Equipment Shutdown Following a Trip
- Reactor Coolant System Mid Loop Operations

TABLE 13.5-2
SYSTEM OPERATING PROCEDURES

(Sheet 1 of 3)

- Reactor Coolant System
- Residual Heat Removal System
- Chemical and Volume Control System
- Reactor Make-up and Chemical Control System
- Boron Thermal Regeneration System
- Concentration Boric Acid System
- Reactor Coolant Pump
- Pressurizer Relief Tank
- Safety Injection System
- Safety Injection Accumulators
- Containment Spray System
- Hydrogen Purge Supply and Exhaust System

- Main Steam System
- Feedwater System
- Condensate System
- Auxiliary Feedwater System
- Steam Generator Blowdown and Cleanup System
- Condensate Polishing System
- Extraction Steam System
- Heater Drains System
- Condenser Vacuum and Waterbox Priming System
- Circulating Water System
- Auxiliary Steam System
- Chemical Feed System
- Turbine Plant Cooling Water System
- Steam Generator Recirculation System
- High Head Chemical Injection System
- Condensate and Feedwater System Long Term Wet Layup and Recirculation
- Turbine Control Fluid System
- Turbine Gland Steam System

TABLE 13.5-2
SYSTEM OPERATING PROCEDURES

(Sheet 2 of 3)

- Turbine Oil Purification System
- Turbine Lube Oil System
- Main Generator System
- Generator Hydrogen System
- Generator Seal Oil System
- Generator Primary Water System
- Station Service Water System
- Component Cooling Water System
- Service Water Chlorination System
- Low Volume Waste
- Spent Fuel Pool Cooling and Cleanup System
- Demineralized and Reactor Make-up Water System
- Service Air System
- Instrument Air System
- Vents and Drains System
- Nitrogen Gas System
- Hydrogen Gas System
- Carbon Dioxide Gas System
- Oxygen Gas System
- 138 KV and 345 KV Startup Transformers
- 6900 v Switchgear
- 480 V Switchgear Motor Control Centers and Power Panels
- 480 v Switchgear and Motor Control Centers
- 125 v DC Switchgear and Distribution System, and Batteries and Battery Chargers (Safety-Related)
- 24/48 v & 125/250 v DC Switchgear and Dist. Systems. Batteries and Battery Chargers (Non-Safety-Related)
- 118 v AC Distribution System and Inverters
- 208 v/120 v AC Distribution System
- Diesel Generator System
- Diesel Generator Fuel Oil Storage and Transfer System

TABLE 13.5-2
SYSTEM OPERATING PROCEDURES

(Sheet 3 of 3)

- Isolated Phase Bus Duct Cooling System
- 345 KV Switchyard Relay House
- Rod Control System
- Excore Instrumentation System
- Area Radiation Monitoring Systems
- Process and Effluent Radiation Monitoring Systems
- Incore Instrumentation System
- Containment Ventilation System
- Control Room Ventilation System
- Switchgear Area Ventilation System
- Battery Room Ventilation Systems
- Auxiliary Building Ventilation System
- Main Steam and Feedwater Area and Electrical Area Ventilation System
- Fuel Building Ventilation System
- Diesel Generator Rooms Ventilation System
- Uncontrolled Access Area Ventilation System
- Office and Service Areas Ventilation System
- Service Water Intake Ventilation System
- Ventilation Chilled Water System
- Safety Chilled Water System
- Primary Plant Ventilation System
- Safeguards Building Ventilation System
- Squaw Creek Reservoir Makeup System
- Squaw Creek Reservoir Return and Service Outlet System
- Halon Fire Protection Systems
- Water Fire Protection System
- Containment Personnel Airlocks

TABLE 13.5-3
ABNORMAL CONDITIONS PROCEDURES

(Sheet 1 of 2)

- REACTOR COOLANT PUMP TRIP/MALFUNCTION
- HIGH REACTOR COOLANT ACTIVITY
- EXCESSIVE REACTOR COOLANT LEAKAGE
- RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION
- CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTIONS
- HIGH SECONDARY ACTIVITY
- EMERGENCY BORATION
- UPS HVAC MALFUNCTION
- LOSS OF NON-SAFEGUARDS VENTILATION SYSTEMS
- INSTRUMENT AIR SYSTEM MALFUNCTION
- FEEDWATER, CONDENSATE, HEATER DRAIN SYSTEM MALFUNCTION
- AUXILIARY FEEDWATER SYSTEM MALFUNCTION
- MAIN TURBINE MALFUNCTION
- MAIN GENERATOR MALFUNCTION
- STATION SERVICE WATER SYSTEM MALFUNCTION
- COMPONENT COOLING WATER SYSTEM MALFUNCTIONS
- RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION
- RESPONSE TO A 6900/480 V SYSTEM MALFUNCTION
- LOSS OF PROTECTION AND/OR INSTRUMENT BUS
- SOURCE RANGE INSTRUMENTATION MALFUNCTION
- INTERMEDIATE RANGE INSTRUMENTATION MALFUNCTION
- POWER RANGE INSTRUMENTATION MALFUNCTION
- Tc/N-16 INSTRUMENTATION MALFUNCTION
- PRESSURIZER PRESSURE INSTRUMENTATION MALFUNCTION
- PRESSURIZER LEVEL INSTRUMENTATION MALFUNCTION
- STEAM FLOW INSTRUMENTATION MALFUNCTION
- FEEDWATER FLOW INSTRUMENTATION MALFUNCTION
- STEAM LINE PRESSURE, STEAM HEADER PRESSURE, AND TURBINE 1st-STAGE PRESSURE INSTRUMENT MALFUNCTION
- STEAM GENERATOR LEVEL INSTRUMENTATION MALFUNCTION
- ROD CONTROL SYSTEM MALFUNCTION

TABLE 13.5-3
ABNORMAL CONDITIONS PROCEDURES

(Sheet 2 of 2)

- RCS FLOW INSTRUMENTATION MALFUNCTION
- WIDE RANGE Tc OR TH INSTRUMENTATION MALFUNCTION
- RCS LOOP PRESSURE INSTRUMENTATION MALFUNCTION
- RESPONSE TO A FIRE IN THE CONTROL ROOM OR CABLE SPREADING ROOM
- RESPONSE TO FIRE IN THE SAFEGUARDS BUILDING
- RESPONSE TO FIRE IN THE AUXILIARY BUILDING OR THE FUEL BUILDING
- RESPONSE TO FIRE IN THE ELECTRICAL AND CONTROL BUILDING
- RESPONSE TO FIRE IN THE CONTAINMENT BUILDING
- RESPONSE TO FIRE IN SERVICE WATER INTAKE STRUCTURE
- RESPONSE TO FIRE IN THE TURBINE BUILDING
- FIRE PROTECTION SYSTEM MALFUNCTION
- ACCIDENTAL RELEASE OF RADIOACTIVE GAS
- ACCIDENTAL RELEASE OF RADIOACTIVE LIQUID
- LOSS OF CONTROL ROOM HABITABILITY
- PLANT PROCESS COMPUTER SYSTEM MALFUNCTION
- ACTS OF NATURE
- FUEL HANDLING ACCIDENT
- SPENT FUEL POOL/REFUELING CAVITY LEAKAGE
- LOOSE PARTS MONITORING
- SHUTDOWN LOSS OF COOLANT
- CONTROL ROOM VENTILATION SYSTEM MALFUNCTION
- MAIN CONDENSER AND CIRCULATING WATER SYSTEM MALFUNCTION
- TURBINE TRIP RESPONSE
- RAPID POWER REDUCTION
- LOSS OF NON 1E INSTRUMENT BUS
- 25 Kv MALFUNCTION
- CONTROL ROOM ANNUNCIATOR SYSTEM AND STATUS LIGHT MALFUNCTION
- COLD WEATHER PREPARATION/HEAT TRACING AND FREEZE PROTECTION SYSTEM MALFUNCTION

TABLE 13.5-4
EMERGENCY OPERATING PROCEDURES

(Sheet 1 of 2)

- Reactor Trip or Safety Injection
- Loss of Reactor Coolant or Secondary Coolant
- Faulted Steam Generator Isolation
- Steam Generator Tube Rupture
- Rediagnosis
- Reactor Trip Response
- Natural Circulation Cooldown
- Natural Circulation cooldown with Steam Void in vessel (with RVLIS)
- Natural circulation cooldown with Steam Void in vessel (without RVLIS)
- Safety Injection Termination
- Post LOCA cooldown and depressurization
- Transfer to cold leg recirculation
- Transfer to hot leg recirculation
- Post-SGTR cooldown using backfill
- Post-SGTR cooldown using blowdown
- Post-SGTR cooldown using Steam Dump
- Loss of all ac power
- Loss of all ac power recovery without SI required
- Loss of all ac power recovery with SI required
- Loss of emergency coolant recirculation
- LOCA outside containment
- Uncontrolled depressurization of all steam generators
- SGTR with loss of reactor coolant - subcooled recovery
- SGTR with loss of reactor coolant saturated recovery desired

TABLE 13.5-4
EMERGENCY OPERATING PROCEDURES

(Sheet 2 of 2)

- SGTR without pressurizer pressure control
- Response to Nuclear Power Generation/ATWT
- Response to Loss of Core Shutdown
- Response to Inadequate Core Cooling
- Response to Degraded Core Cooling
- Response to Saturated Core Cooling
- Response to Loss of Secondary Heat Sink
- Response to steam generator overpressure
- Response to steam generator high level
- Response to Loss of Normal Steam Release Capabilities
- Response to Steam Generator Low Level
- Response to Imminent Pressurized Thermal Shock Condition
- Response to Anticipated Pressurized Thermal Shock Condition
- Response to High Containment Pressure
- Response to Containment Flooding
- Response to High Containment Radiation Level
- Response to High Pressurizer Level
- Response to Low Pressurizer Level
- Response to Voids in Reactor Vessel

TABLE 13.5-5
RADWASTE SYSTEMS PROCEDURES

- Reactor Coolant Drain Tank Subsystem
- Drain Channel A
- Drain Channel B
- Drain Channel C
- Boron Recycle System
- Gaseous Waste Processing System
- Radwaste Solidification System
- Spent Resin Handling System
- Spent Cartridge Filter Removal and Disposal

TABLE 13.5-6
PLANT RADIATION PROTECTION ACTIVITIES

- Surveying and Monitoring to Evaluate Radiation Hazards
- Health Physics Indoctrination and Training
- Ingress/Egress Requirements for Restricted Areas
- Use and Maintenance of Protective Equipment
- Recording, Storing and Reporting of Occupational Radiation Exposures
- Use, Maintenance, and Calibration of Fixed and Portable Health Physics Instrumentation
- Personnel, Equipment and Area Decontamination
- Control of Personnel, Equipment, and Areas to Mitigate the Spread of Radioactive Contamination
- Radiation Work Permits
- Use, Maintenance and Calibration of the Radiation Monitoring System
- Receiving, Packaging and Shipping of Radioactive Material

TABLE 13.5-7
TABLE 13.5-7 HAS BEEN DELETED.

TABLE 13.5-8
CHEMICAL/RADIOCHEMICAL CONTROL

- Approved Techniques for Obtaining Liquid, and Gaseous Samples
- Sampling Schedules for Primary, Secondary, and Associated System
- Recording, Notification, and Retention of Chemical/Radiochemical Data
- Chemical Analysis of Primary, Secondary, and Associated System Samples
- Radiochemical Analysis of Samples
- Proper Use, Maintenance, and Calibration of Portable and Fixed Chemical/ Radiochemical Instruments
- Maintaining Primary, Secondary, and Associated System Water Quality
- Proper Use and Preparation of Standard Chemical/Radiochemical Solutions
- Chemical Additive Systems
- Sampling and Releasing of Liquid and Gaseous Radioactive Waste

TABLE 13.5-9
OPERATIONS DEPARTMENT ADMINISTRATION PROCEDURES

- Operations Department Organization and Responsibilities
- Shift Complement, Responsibilities and Authorities
- Operations Department Document Control
- Equipment Custody Transfer From Startup to Operations
- Review of Documents
- Reporting of Operational Incidents
- Post Trip Review Evaluation
- Human Engineering Review
- Preparation of Integrated Plant Operating Procedures
- Preparation of System Operating Procedures
- Preparation of Abnormal Condition Procedures
- Preparation of Emergency Operating Procedures
- Preparation of Alarm Procedures
- Preparation of Operations Test Procedures and Equipment Test Procedures
- Guidelines for the Preparation and Review of Operations Procedures
- Preparation and Control of Technical Data
- Operations Department Work Instructions
- Operating Logs
- Relief of Personnel
- Conduct of Personnel in Control Room
- LCO Tracking Log
- Operations Department Physicals
- Disabling of Control Panel Annunciators/Instruments
- Operations Department Key Controls>P
- Guidelines on Component Position Verification
- Guideline on use of Procedures
- Non-Standard Alignments and Evolutions
- Operations Control of Lubricants

13.6 SECURITY

A comprehensive security program has been developed at CPNPP in accordance with the requirements of applicable portions of 10CFR73, “Physical Protection of Plants and Materials.” The security plans that describe the program contain security related information (SRI) or safeguards information (SGI) and are withheld from public disclosure in accordance with 10CFR2.390 or 10CFR73.21. The CPNPP security plans are presented in separate submittals.

14.0 INITIAL TEST PROGRAM

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14.0 INITIAL TEST PROGRAM

LIST OF TABLES

Number	Title
14.2-1	NOT USED
14.2-2	PREOPERATIONAL TESTS SUMMARIES INDEX
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14.0 INITIAL TEST PROGRAM

LIST OF FIGURES

Number	Title
14.2-1	Test Review Group
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14.2-3	Preoperational Test Schedule (2 Sheets)
14.2-4A	Initial Startup Test Schedule (Unit 1)
14.2-4B	Initial Startup Test Schedule (Unit 2)

14.1 SPECIFIC INFORMATION TO BE INCLUDED IN PRELIMINARY SAFETY ANALYSIS REPORTS

This section is not applicable; see Section 14.2 for specific information.

14.2 INITIAL TEST PROGRAM

14.2.1 SUMMARY OF TEST PROGRAM AND OBJECTIVES

The Initial Test Program for CPNPP Units 1 and 2 has been completed. The activities described in Section 14.2 for the preoperational and initial startup test programs will be maintained as historical information and will not be revised to reflect updates to organizational structures or position titles and responsibilities. Those types of changes will be reflected in Chapters 13 and 17, as applicable.

The purpose of the test program for Comanche Peak Nuclear Power Plant (CPNPP) is to assure that the installed station structures, systems and components will be subjected to tests as required to verify that the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public, and to provide assurance of total plant reliability for operation. The test program will also ensure that the procedures for operating the plant safely have been evaluated and demonstrated, and that the operating organization is knowledgeable about the plant and procedures and is prepared to safely operate the facility.

The necessary procedures to control, implement, and document the test program are established by startup and plant administrative procedures and summarized in the following sections. TU Electric, as the applicant, has responsibility for overall direction and management of the test program.

The testing activities to be performed on safety-related systems at the CPNPP are divided into two major phases: preoperational testing and initial startup testing.

The preoperational testing phase is divided into two overlapping parts: prerequisite testing and preoperational testing. Prerequisite testing will be conducted in order to verify the integrity, proper installation, cleanliness, and functional operability of the system components. Preoperational testing will be performed in order to demonstrate the capability of systems, structures, and components perform their safety functions.

These structures, systems, and components are identified as quality related items in Table 17A-1 of FSAR Section 17A. In addition, systems and components which have applicable testing requirements specified in Regulatory Guide 1.68 and discussed in Appendix 1A(B) will be subjected to preoperational tests. Summaries of individual preoperational tests are provided in Section 14.2.12.

Preoperational testing will be completed prior to fuel load with certain limited exceptions where tests or parts of tests will be deferred to the Initial Startup Test Program. Such an exception is the complete testing of the control rod drive mechanisms with the rods attached. In such cases sufficient tests will be performed prior to fuel loading to provide reasonable assurance that the post-loading tests will be successful.

Initial startup tests will be performed beginning with activities leading to fuel loading and ending with full power operation. The intent of these tests is to assure that tests deferred from the preoperational test program are performed; that fuel loading is effected in a safe manner; that the plant is safely brought to rated capacity; that plant performance is satisfactory in terms of established design criteria; and to demonstrate, where practical, that the plant is capable of

withstanding anticipated transients and postulated accidents. Testing activities related to fuel load and initial criticality are described in Section 14.2.10. Initial startup tests including low power physics testing and power ascension testing are described in the test summaries provided in Section 14.2.12.

For systems and components which have no nuclear safety-related requirements, acceptance testing will be performed to verify proper system performance and to ensure reliable and efficient operation of the plant.

Preoperational and initial startup tests will be conducted in accordance with approved test procedures. Review, approval, and revision of test procedures and the evaluation and disposition of test results will be accomplished by methods specified in the appropriate administrative procedures and summarized in Sections 14.2.3 and 14.2.5.

The startup program will utilize, to the extent practical, operations personnel and operating procedures to provide familiarization with the plant installation and demonstrate the adequacy of operating procedures.

Preoperational testing activities will be coordinated through a Joint Test Group (JTG), as described in Section 14.2.2.6. Initial startup testing activities will be coordinated through a Test Review Group (TRG), as described in Section 14.2.2.7.

14.2.2 ORGANIZATION AND STAFFING

14.2.2.1 General Description

TU Electric is responsible for the overall administration and technical direction of the CPNPP test program. The Performance and Test Manager is responsible for the coordination of the initial startup test program. The Manager, Startup is responsible for the coordination, direction, and implementation of the prerequisite and preoperational test program.

The TRG and JTG organizations are shown in Figures 14.2-1 and 14.2-2. The SORC membership is described in Section 13.4.1.1.1.

14.2.2.2 Performance and Test Organizational Responsibilities

The duties and responsibilities of the Performance and Test organization under the direction of the Performance and Test Manager are as follows:

1. Preparation of testing schedules;

2. Coordination with the appropriate engineering organizations to resolve component/ system design and operating problems;

3. Preparation of test procedures;

4. Conduct tests to demonstrate safe plant operation;

5. Perform the initial startup test sequence to ensure a safe and orderly power ascension program.

Support personnel including craft labor and technicians who will be engaged in testing activities may be supplied by TU Electric or contractors. Such support will vary according to test program requirements.

14.2.2.2.1 Performance and Test Manager

The responsibilities of the Performance and Test Manager include:

1. Development and implementation of the Performance and Test surveillance tests;

2. Development and implementation of the Initial Startup Test Program;

3. Coordination of testing with the activities of operations, maintenance, engineering, and construction organizations;

4. Administration of the development of plans and schedules to support the assigned testing activities;

5. Administration of the development of individual test procedures;

6. Review test procedures and results or assure that such reviews are conducted by qualified personnel within the Performance and Test organization;

7. Assure proper and timely notifications and reports pertaining to assigned testing activities are submitted to the Nuclear Regulatory Commission and other regulatory agencies;

8. Implement the program to utilize testing and operating experience in other similar nuclear plants.

14.2.2.2.2 Performance and Test Engineers

The duties and responsibilities of the Performance and Test engineers include:

1. Preparation of assigned test procedures to direct and guide specific tests in accordance with a standard format;

2. Direction to support personnel and others during performance of tests including appropriate interface with station operators;

3. Ensuring the safety of personnel and plant equipment during testing;

4. Familiarization of support personnel with specific tests;

5. Identification of deficiencies that could adversely affect test performance;

6. Assembly of test data and preparation of test reports for evaluation of test results by others; and

7. Responsibility to disallow or terminate testing due to conditions which could endanger personnel or equipment.

14.2.2.3 Startup Department Organizational Responsibilities

The duties and responsibilities of the Startup Department under the direction of the Manager, Startup are as follows:

1. Preparation of testing schedules;

2. Coordination with appropriate construction organizations for release of completed components/systems to facilitate testing;

3. Coordination with the appropriate engineering organizations to resolve component/ system design and operating problems;

4. Preparation of test procedures;

5. Conduct tests to demonstrate adequate and safe component and system performance.

6. Administration of the preventive and corrective maintenance program for Unit 2 equipment under project jurisdiction.

Support personnel including craft labor and technicians who will be engaged in testing activities may be supplied by TU Electric or contractors. Such support will vary according to test program requirements.

14.2.2.3.1 Manager, Startup

The responsibilities of the Manager, Startup include:

1. Administration of the development of plans and schedules regarding the status of the startup program;

2. Administration of the development of individual test procedures;

3. Continuing analysis of construction and equipment installation schedules for compatibility with testing schedules and implementation of corrective actions to minimize conflicts;

4. Review test procedures and results or assure that such reviews are conducted by qualified personnel within the Startup Department;

5. Assure proper and timely notifications and reports pertaining to testing activities are submitted to the Nuclear Regulatory commission and other regulatory agencies.

14.2.2.3.2 Assistant Startup Managers

The Assistant Startup Managers are responsible for the startup program and will have the following duties and responsibilities:

1. Provide technical direction to system test engineers and others assigned to the preoperational test group;

2. Review of Startup Department administrative and prerequisite test procedures and ensure that the procedures and test results have been approved at the appropriate level;

3. Review and submittal of design change requests originated by the Startup Department in accordance with the appropriate Startup Department administrative procedures;

4. Assure testing activities are conducted in accordance with the Quality Assurance Program and applicable procedures; and

5. Coordination of testing with the activities of the engineering, construction and operations organizations.

14.2.2.3.3 Startup Engineers

The duties and responsibilities of the Startup engineers include:

1. Preparation of assigned test procedures to direct and guide specific tests;

2. Direction to support personnel and others during performance of tests including appropriate interface with station operators;

3. Ensuring the safety of personnel and plant equipment during testing;

4. Familiarization of support personnel with specific test;

5. Identification of deficiencies that could adversely affect test performance;

6. Assembly of test data and preparation of test reports for evaluation of test results by others;

7. Implementation of tagging procedures; and

8. Responsibility to disallow or terminate testing due to conditions which could endanger personnel or equipment.

14.2.2.4 Operating Staff

The CPNPP operating staff will be involved in the test program in several capacities throughout preoperational and initial startup testing. This involvement will include review of test procedures and results and the direct participation of operating personnel in test activities. Operating staff technicians will be assigned to assist test engineers in performing tests. Station operators will assist test engineers in performing tests and will take over the routine operations of systems for which prerequisite-type testing has been completed. The operating staff will direct the fuel loading and will be responsible for the operation of the plant during initial startup testing.

The duties and responsibilities of the operating staff during plant operation are described in Chapter 13. The duties of key operating personnel with regard to the startup program are summarized below.

14.2.2.4.1 Vice President, Nuclear Operations

The Vice President, Nuclear Operations has overall responsibility for station operations and the test program.

14.2.2.4.2 Plant Manager

The Plant Manager will provide a continual analysis of testing schedules, operator training schedules, and plant operating staff workloads in order to minimize conflicts with the test program, and he also will provide coordination between the operating staff and the Startup Department.

14.2.2.4.3 Operations Department

The Manager, Operations will be responsible for the proper operation of all equipment in the custody of the operations department and for ensuring that the conduct of the test program does not place the plant in an unsafe condition. He will provide personnel from the operating staff as required to support the conduct of testing activities. In addition, he will direct the development of station operating procedures.

The Shift Supervisors report to the Manager, Operations and are responsible for the safe operation of the plant during assigned shifts. They also are responsible for the implementation of appropriate safety and custody tagging procedures and have responsibility to disallow or terminate testing due to conditions which could endanger personnel or equipment or violate Technical Specifications.

14.2.2.4.4 Maintenance Department

The Manager, Maintenance will be responsible for performing preventive and corrective maintenance when required on Unit 2 components and systems that have been released to operations from the startup department. He will provide personnel from the maintenance department to support testing activities as required.

14.2.2.4.5 Plant Engineering

The Manager, Plant Engineering will provide personnel from the Plant Engineering organization as required to support testing activities. The Performance and Test Manager reports to the Manager, Plant Engineering and is responsible for the implementation of the Initial Startup program.

14.2.2.4.6 Director, Oversight & Regulatory Affairs

The Director, Oversight & Regulatory Affairs is responsible for approving and verifying implementation of the TU Electric CPNPP QA Program.

14.2.2.5 Major Participating Organizations

14.2.2.5.1 Design Engineering

Design Engineering will be responsible for assisting to the extent required in ensuring that tests sufficiently verify the adequacy of system design and in the formulation of changes required to correct detected design deficiencies. This responsibility includes:

1. Review of preoperational and initial startup test objectives and acceptance criteria to verify implementation of operating license commitments and consistency with the station safety analyses;

2. Review and approval of design change requests originated during the course of the test program.

3. Management of contractors.

14.2.2.5.2 Brown & Root, Inc. (B&R)

B&R, as the constructor for CPNPP, is responsible for the construction completion, performance of associated constructions tests, and orderly release of components and systems to TU Electric consistent with the test program schedules. This responsibility includes:

1. Completion of construction and construction testing activities;

2. Provision of craft technical manpower support as required for performance of the test program.

14.2.2.5.3 Westinghouse Electric Corporation

Westinghouse, as the Nuclear Steam Supply System (NSSS) supplier, is responsible for providing technical direction to TU Electric during preoperational and initial startup testing performed on the NSSS and associated auxiliary equipment. Technical direction is defined as technical guidance, advice and counsel based on current engineering, installation, and testing practices. This responsibility includes:

1. Assignment of personnel to provide advice and assistance to TU Electric for test and operation of all equipment and systems in the Westinghouse area of responsibility.

2. Assignment of an operational physicist to the site organization during fuel loading and low power testing; and

3. Provision of test procedure outlines and technical assistance for tests of Westinghouse furnished components and systems.

14.2.2.5.4 Utility Power Corporation (UPC)

UPC, as supplier of the main turbine generator set, is responsible for providing technical support to TU Electric during preoperational and initial startup testing performed on the turbine generator and related auxiliary equipment.

14.2.2.6 Joint Test Group (JTG)

The JTG is comprised of certain station supervisory and technical personnel, is charged with reviewing Startup activities and advises the Manager, Startup on the disposition of those items reviewed.

14.2.2.6.1 JTG Membership

As a minimum, the JTG shall be composed of managers, supervisors, or engineers from the organizations listed below.

Startup

Operations

Engineering (Design & Technical Support functions)

Westinghouse (for all matters concerning preoperational testing performed on the NSSS and associated auxiliary systems)

Quality Assurance

The members and their alternates, including the Chairman, shall be designated in writing by their cognizant Vice President.

The primary function of the JTG is the review and approval of all preoperational test procedures, procedure revisions, and test results.

In addition to the above, representatives of other organizations will participate, as requested by the Chairman.

14.2.2.7 Test Review Group (TRG)

The TRG is comprised of certain station supervisory and technical personnel. The TRG functions as a subcommittee of the Station Operations Review Committee (SORC) for initial startup testing matters. The TRG is charged with reviewing initial startup testing activities and advising the SORC on the disposition of those items reviewed. The SORC may act in lieu of the TRG for activities described below.

14.2.2.7.1 TRG Membership

As a minimum, the TRG shall be composed of management, supervisory or technical representatives from the areas listed below.

- Operations
- Engineering (design, testing and technical support functions)
- Westinghouse (for all matters concerning initial startup testing performed on the NSSS and associated auxiliary systems)

The members and their alternates, including the Chairman, shall be designated in writing by the Vice President, Nuclear Operations.

The primary function of the TRG is the review and approval of initial startup program test procedures, procedure revisions, and test results.

In addition to the above, representatives of other organizations will participate, as requested by the Chairman.

14.2.2.8 Nuclear Oversight

The Luminant Power Nuclear Oversight Department is responsible for assuring the quality of construction, plant testing, and operations activities as described in Chapter 17.

14.2.2.9 Qualifications of Key Personnel

The minimum qualifications of individuals responsible for review and approval of preoperational and initial startup test procedures and results shall be; (1) Bachelor's degree in engineering or the physical sciences and four years of applicable power plant experience; or (2) high school diploma or equivalent and eight years of applicable power plant experience. Credit for up to two years for related technical training may be substituted on a one-for-one time basis. A minimum of two years shall be applicable nuclear power plant experience.

The minimum qualifications of individual that direct or supervise the conduct of preoperational test at the time of performance of the duties, shall be; (1) Bachelor's degree in engineering or the physical sciences and one year of experience in power plant testing or operation. Included in the one year shall be three months of familiarization of system and component operation unique to the design of similar nuclear power plants or (2) a high school diploma or equivalent and four years of experience in power plant testing or operation or credit for up to two years for related technical training may be substituted on a one-for-one time basis. Included in the four years shall be three months of familiarization of system and component operation unique to the design of similar nuclear power plants.

The minimum qualifications of individuals that direct or supervise the conduct of initial startup test at the time of performance of the duties, shall be: (1) Bachelor's degree in engineering or the physical sciences and two years of experience in power plant testing or operation. Included in the two years shall be a minimum of one year of nuclear power plant testing, operation or training on a nuclear facility, or (2) a high school diploma or equivalent and five years of experience in power plant testing of which two years will be nuclear power plant experience. Credit for up to two years for related technical training may be substituted on a one-for-one time basis.

Test personnel will be indoctrinated in the use of applicable administrative procedures, test procedures and familiarized with applicable quality assurance requirements.

14.2.3 TEST PROCEDURES

14.2.3.1 General

Preoperational and initial startup tests will be performed in accordance with written approved test procedures. The following sections describe the general methods employed to control procedure

development and review, and they also describe the responsibilities of the various organizational units participating in this process.

The detailed controls and methods will be prescribed in the startup administrative procedures and station administrative procedures, as applicable.

14.2.3.2 Development of Procedures

Technical information required for the preparation of the test procedures will be provided by the appropriate engineering organizations. This information will consist of system descriptions, technical specifications, design drawings and other technical documents which define the functional requirements and performance objectives for the various systems and components. Additional technical data may also be obtained from the various component vendors and other contractors as required.

The applicable functional requirements provided by the system designers will be incorporated into the acceptance criteria for each test procedure. This information will also be used by the test engineer in developing the detailed test methods which ensure that the capability of systems and components to function properly within design specifications is adequately demonstrated.

14.2.3.3 Review and Approval of Test Procedures

Preoperational and initial test procedures will be reviewed by members of the plant operating staff and appropriate design organizations.

Preoperational test procedures will be forwarded to appropriate members of the Joint Test Group for review and comment.

Initial startup test procedure will be forwarded to the appropriate members of the Test Review Group for review and comment.

The responsibility for final approval of preoperational test procedures rests with the Manager, Startup. The responsibility for final approval of initial startup test procedures rests with the Plant Manager.

14.2.3.4 Format of Test Procedures

Preoperational and initial startup tests will be prepared based on formats specified by administrative procedures. These standard formats will help ensure that each procedure contains the information and instructions required to satisfactorily perform and document the test.

The procedures format and content will reflect the guidance provided in Regulatory Guide 1.68 as discussed in Appendix 1A(B). The standard format will include, as a minimum, the following:

1. Test Objectives:

   A detailed statement of the test objectives and method of system or plant operation to be demonstrated.

2. References:

   References to technical specifications, supporting procedures, vendor's manual, or other technical documents will be included as required.

3. Prerequisites:

   Prerequisites and initial conditions are satisfied and the necessary prerequisite tests and construction activities have been satisfactorily completed.

4. Precautions and Limitations:

   Precautions and limitations relating to personnel safety, equipment integrity, and overall plant safety will be specified.

5. Instructions:

   The instruction section will contain detailed step by step instructions for operating the system in the test configuration, performing actual test manipulations, and for use of off-normal procedures such as jumper cables or mechanical bypasses and restoration of the system to normal status following test. The detailed test instructions will utilize normal and emergency station operating procedures to the extent practical.

6. Acceptance Criteria:

   The performance objectives and functional requirements for system operations will be specified. The criteria used to judge the success or failure of the test may be qualitative or quantitative.

7. Data Collection:

   Provisions will be made for recording all pertinent test data regarding system conditions and performance. Records will identify the observer, type of test instrumentation used, acceptability of results, and any deficiencies, and will become a part of the permanent station records.

14.2.3.5 Revisions to Procedures

Revisions to preoperational test procedures will be reviewed by appropriate members of the JTG and approved by the Manager, Startup. Revisions to initial startup test procedures will be reviewed by the appropriate members of the TRG and approved by the Plant Manager.

Preoperational and Initial Startup Test procedure modifications required during conduct of test will be approved as follows;

1. Preoperational Test Procedures

   a. Test procedure modifications that do not change the intent of the test will be approved by the system test engineer.

b. Test procedure modifications that change the intent of the test will be approved by the test engineer and another member of the startup organization qualified to review preoperational test procedures.

c. All procedure modifications will be included with the completed procedure and be subject to review and approval with the test results as described by paragraph 14.2.5.

2. Initial Startup Test Procedures

a. Modifications to initial startup tests that do not change the intent of the test will have the Shift Supervisor's concurrence and approval of the person in charge of the test.

b. Modifications to initial startup tests that change the intent of the test will be reviewed and approved by the Plant Manager.

14.2.4 CONDUCT OF TEST PROGRAM

14.2.4.1 Administrative Procedures

The conduct of the test program will be controlled by administrative procedures. These procedures will prescribe controls for testing activities such as the following:

1. Preparation, review and approval, of test procedures

2. Turnover of systems

3. Format and content of test procedures

4. Safety and custody tagging procedures

5. Temporary system modifications

6. Design change processing

7. Test deficiency processing

8. Review of Reactor Operating/Startup experiences.

14.2.4.2 Prerequisite Testing

Startup administrative procedures will be established to ensure that applicable prerequisites are met before testing is initiated. Upon completion of construction phase activities, custody of the component/system will be normally transferred to TU Electric for conduct of prerequisite testing and preoperational testing. In select cases prerequisite testing may be conducted prior to custody transfer with agreement of construction and startup. During the turnover process, systems and components will be reviewed for completeness, installation damage and conformance with appropriate installation and/or design documents. Outstanding construction, document and test deficiencies will be identified and controlled prior to fuel load.

14.2.4.3 Preoperational Testing

Technical direction and administration, including test procedure preparation, test execution, and data recording, of the preoperational testing is the responsibility of the Startup Department, with the operating staff retaining responsibility for performing actual equipment operations and maintenance.

The Manager, Startup is responsible for the administration and implementation of all preoperational testing activities during the startup program.

The test engineers will direct support personnel in the performance of tests and will provide appropriate interface with station operators. The Shift Supervisors will be responsible for insuring that the conduct of testing does not place the plant in an unsafe condition at any time. Additionally, the shift supervisors and test engineers have the authority to terminate or disallow testing at any time.

14.2.4.4 Initial Startup Testing

Technical direction and administration, including test procedure preparation, test execution, and data recording, of initial startup testing is the responsibility of the Performance and Test organization, with the operating staff retaining responsibility for performing actual equipment operations and maintenance.

The Performance and Test Manager is responsible for the administration and implementation of all initial startup testing activities during the test program.

The test engineers will direct support personnel in the performance of tests and will provide appropriate interface with station operators. The Shift Supervisors will be responsible for insuring that the conduct of testing does not place the plant in an unsafe condition at any time. Additionally, the Shift Supervisors, other on-shift licensed operators, and test engineers have the authority to terminate or disallow testing at any time.

14.2.4.5 Test Prerequisites

Each test procedure will contain a set of prerequisites or initial conditions as prescribed by administrative procedures. The test engineer will ensure that all specified prerequisites are met prior to performing the test. The format for test procedures is described in Section 14.2.3.4.

14.2.4.6 Phase Evaluation

The test results of all preoperational tests performed in the preoperational testing phase will be reviewed by the JTG prior to commencing the initial startup testing phase of the initial test program.

Between each major phase of the initial startup test program the test results for all tests that have been performed will be reviewed by the TRG.

This review ensures that all required systems have been tested satisfactorily and that test results have been evaluated before proceeding to the next stage of testing. This review is described in Section 14.2.5.

14.2.4.7 Design Modifications

Modifications to the design of safety-related equipment during the test program may be initiated in order to correct deficiencies discovered as a result of testing. Any such modification will be referred to the appropriate engineering organization for approval. Modifications made to safety-related components or systems after completion of preoperational or initial startup testing will be reviewed for retesting requirements on affected portions of the system.

14.2.5 REVIEW, EVALUATION, AND APPROVAL OF TEST RESULTS

Following completion of a particular test, a test engineer will assemble the test data package for evaluation. Preoperational tests will be reviewed by appropriate members of the JTG. Initial startup test results will be reviewed by appropriate members of the TRG.

Each test data package will be reviewed to ensure that the test has been performed in accordance with the written approved procedure and that all required data, checks, and signatures have been properly recorded and that system performance meets the approved acceptance criteria.

Deficiencies identified in the review process will be resolved to the satisfaction of the appropriate review group. If the evaluation indicates that deficiencies in the test method are responsible for unsatisfactory test results, the test procedure will be modified accordingly before retesting is initiated. Whenever an evaluation of test results indicates deficiencies in system performance, the problem will be referred to the appropriate engineering organization for evaluation.

The responsibility for final approval of preoperational test results rests with the Manager, Startup. The responsibility for final approval of initial startup test results and authorization to proceed to higher power level following completion of major power plateau testing (i.e., 0-5%, 50% and 75%) rests with the Plant Manager.

Following each major phase of the test program, test results and/or test status will be reviewed to ensure that all required tests have been performed and that the test results have been evaluated. This evaluation will ensure that all required systems are operating properly and that testing for the next major phase will be conducted in a safe and efficient manner. This type of review will be performed to the extent required before major test phases such as fuel load, initial criticality, and power escalation. During the power escalation phase, review and evaluation of initial startup test procedure results will be completed for each major power plateau prior to proceeding with power ascension testing to the next plateau.

14.2.6 TEST RECORDS

Test procedures and test data relating to preoperational and initial startup testing will be retained in accordance with the measure described in Section 17.2.17, “Quality Assurance Records.”

14.2.7 CONFORMANCE OF TEST PROGRAMS WITH REGULATORY GUIDES

The test program will be conducted in accordance with guidance given in the applicable Nuclear Regulatory Commission Regulatory Guides which are listed below.

1. Regulatory Guide 1.41, “Preoperational Testing of Redundant On- Site Electric Power System to Verify Proper Load Group Assignments”

2. Regulatory Guide 1.52, “Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants”

3. Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants”

4. Regulatory Guide 1.68.2, “Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants.”

5. Regulatory Guide 1,69, “Concrete Radiation Shields for Nuclear Power Plants.”

6. Regulatory Guide 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing.”

7. Regulatory Guide 1.79, “Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors.”

Exception has been taken to certain provisions of the above Regulatory Guides and are discussed in Appendix 1A(B)

14.2.8 UTILIZATION OF REACTOR OPERATING AND TESTING EXPERIENCE IN DEVELOPMENT OF TEST PROGRAM

The CPNPP test program will utilize information gained from operating and testing experience in other similar nuclear plants. This information will be used to provide guidance in developing test procedures and schedules and to alert personnel to potential problem areas in the testing program.

The development of this program for utilizing testing and operating experience is the responsibility of the Vice President, Nuclear Operations with the direct implementation of the program being the responsibility of the Manager, Startup and the Performance and Test Manager.

Significant event information received from NSAC and INPO will be reviewed by the test organization to identify special testing which should be included in the test program.

The operating experience program will consist of an initial review to be administered prior to conducting preoperational tests and an ongoing review during the remainder of the test program. The initial review will examine all pertinent operating data and abnormal events on similar plants which occurred during a period of two years prior to the review. The review will be conducted so as to allow sufficient time for data to be analyzed and incorporated in test procedures. Any new information will be reviewed on a regular basis during the test program so as to address current testing problems.

Additionally, testing and operating experience gained from Unit 1 will be incorporated into the Unit 2 test program.

14.2.9 TRIAL USE OF PLANT OPERATING AND EMERGENCY PROCEDURES

To the extent practical, the plant operating, emergency, and surveillance procedures will be use-tested during the test program and will also be used in the development of preoperational and initial startup procedures. The trial use of operating procedures serves to familiarize operating personnel with systems and plant operation during the testing phase and also serves to assure the adequacy of the procedures under actual or simulated operating conditions before plant operation begins.

Prior to fuel load, draft operating procedures may be utilized for equipment operation and may be informally altered to meet special test considerations.

In general the development of plant operating procedures will take place in approximately the same time frame as the preparation of preoperational and initial startup test procedures. The operating procedures will be revised as necessary to reflect experience gained during the actual testing.

During the test program, operating procedures employed for periodic tests and inspections may be used directly for certain preoperational and initial startup tests. Operating procedures used directly for preoperational or initial startup tests to obtain quantitative data to be used for evaluation of system performance will be performed in accordance with the CPNPP Station Administrative Manual.

14.2.10 INITIAL FUEL LOADING, CRITICALITY, AND POWER OPERATION

Fuel loading will begin after the required preoperational tests are satisfactorily completed. At the completion of fuel loading, the reactor upper internals and pressure vessel head will be installed and additional mechanical and electrical tests will be performed to prepare the plant for nuclear operation. After final precritical tests, nuclear operation of the reactor will begin. This phase of testing includes initial criticality, low power testing and power level escalation. The purpose of these tests is to establish the operational characteristics of the unit and core, to acquire data for the proper calibration of setpoints, and to ensure that operation is within license requirements. Section 14.2.12.2 summarizes the individual tests which will be performed from core load to rated power. The core loading and post loading tests are described below.

14.2.10.1 Fuel Loading

The overall responsibility for coordination of initial core loading activities will be exercised by the Plant Manager, or his designated representative, with technical assistance provided by Westinghouse. The overall process of initial core loading will be directed from the refueling area of the containment structure by a licensed Senior Reactor Operator.

The core configuration will be specified as part of the core design studies conducted well in advance of fuel loading. In the event mechanical damage is sustained during core loading operations to a fuel assembly, an alternate core loading scheme will be determined. Any such changes will be approved by the appropriate TU Electric and Westinghouse personnel.

Core loading procedures will specify the condition of fluid systems to prevent inadvertent changes in boron concentration of the reactor coolant; the movement of fuel to preclude the

possibility of mechanical damage; the conditions under which loading can proceed; and the responsibility and authority for continuous and complete fuel and core component accountability.

The following conditions will be met prior to core loading:

1. The reactor Containment structure will be complete and Containment integrity established and maintained during fuel loading.

2. Fuel handling tools and equipment will have been checked out and operators familiarized in the use and operation of equipment. Inspections of fuel assemblies, rod cluster control assemblies, and reactor vessel will be satisfactorily completed.

3. The reactor vessel and associated components will be in a state of readiness to receive fuel. Water level will be maintained above the nozzles and recirculation maintained to assure the required boron concentration. Boron concentration can be increased via the recirculation path to the reactor vessel.

Criteria for safe loading require that loading operations stop immediately if any of the following conditions occur.

1. An unanticipated increase in the neutron count rates by a factor of two occurs on all responding nuclear channels during any single loading step after the initial nucleus of eight fuel assemblies is loaded.

2. An unanticipated increase in the count rate by a factor of five on any individual responding nuclear channel during any single loading step after the initial nucleus of eight fuel assemblies is loaded.

3. An unanticipated decrease in boron concentration greater than 20 ppm is determined from two successive samples of the reactor coolant.

Loading operations may not be re-started until the situation is evaluated. An alarm in the Containment and main Control Room will be coupled to the source range channels with a setpoint equal to or less than five times the current count rate. This alarm will automatically alert the loading operation personnel of high count rate and an immediate stop of all operations will be required until the situation is evaluated. In the event the evacuation alarm is actuated during core loading and after it has been determined that no hazards to personnel exist, preselected personnel will be permitted to reenter the Containment to evaluate the cause and determine future action.

The core will be assembled in the reactor vessel, submerged in reactor grade water containing sufficient dissolved boric acid to maintain a calculated core effective multiplication factor of 0.95 or lower. The refueling cavity will be dry during initial core loading with the exception of the refueling canal and upender pits which contain water lubricated equipment. Core moderator chemistry conditions (particularly boron concentration) will be prescribed in the core loading procedure document and verified by chemical analysis of moderator samples taken prior to and during core loading operations.

At least two artificial neutron sources will be introduced into the core at specified points in the core during the loading program to ensure a detector response of at least 1/2 count per second attributable to neutrons.

Core loading instrumentation consists of two permanently installed source range (pulse type) nuclear channels and two temporary incore source range channels. A third temporary channel may also be used as a spare. The permanent channels, when responding, will be monitored in the Main Control Room and the temporary channels will be installed and monitored in the Containment. At least one permanent channel will be equipped with an audible count rate indicator. Both plant channels will have the capability of displaying the neutron flux level on a strip chart recorder. The temporary channels will indicate on scalers. Minimum count rates of 1/2 count per second, attributable to core neutrons will be required on at least two available source range channels at all times following installation of the initial nucleus of eight fuel assemblies. A response check of nuclear instruments to a neutron source will be performed within eight hours prior to loading of the core, or upon resumption of loading if delay is for more than eight hours.

Fuel assemblies together with inserted components (control rod assemblies, burnable poison inserts, source spider, or thimble plugging devices) will be placed in the reactor vessel one at a time according to a previously established and approved sequence developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The core loading procedures document and prescribe the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position in the core. Checks will be made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components. Fuel assembly status boards will be maintained throughout the core loading operation.

An initial nucleus of eight fuel assemblies, one containing a neutron source, is the minimum source-fuel nucleus which will permit subsequent meaningful inverse count rate monitoring. This initial nucleus is determined by calculation to be markedly subcritical (Keff≤0.95) under the required conditions of loading.

Each subsequent fuel addition will be accompanied by detailed neutron count rate monitoring to determine that the just-loaded fuel assembly does not excessively increase the count rate and that the extrapolated inverse count rate ratio is behaving as expected. These results for each loading step will be evaluated before the next fuel assembly is placed in the active core region. The final, as loaded, core configuration will be subcritical (Keff≤0.95) under the required loading conditions.

14.2.10.2 Initial Criticality

Prior to initial criticality, the following tests will be performed and the results evaluated.

1. At the completion of core loading, the reactor upper internals and pressure vessel head will be installed. A pressure test will be conducted after filling, venting and heatup is completed to check the leak tightness of the vessel head installation.

2. Mechanical and electrical tests will be performed on the control rod drive mechanisms. These tests include a complete operational checkout of the mechanisms and calibration checks of the individual rod position indicators.

3. A test will be performed on the reactor trip circuits to test manual trip operation, and actual control rod assembly drop times will be measured for each control rod assembly. At all times that the control rod drive mechanisms are being tested, the boron concentration in the coolant will be maintained such that the shutdown margin requirements specified in the Technical Specifications are met. During individual RCCA or RCC bank motion, source range instrumentation will be monitored for unexpected changes in core reactivity.

4. The Reactor Control and Reactor Protection Systems will be checked with simulated inputs to produce trip signals for various trip conditions.

5. A functional electrical and mechanical check will be made of the incore nuclear flux mapping system near normal operating temperature and pressure.

Initial criticality will be achieved by a combination of shutdown and control bank withdrawal and reactor coolant system boron concentration dilution. The plant conditions, precautions and specific instructions for the approach to criticality will be specified by approved procedures.

Initially, the shutdown and control banks of control rods will be withdrawn incrementally in the normal withdrawal sequence leaving the last withdrawn control bank partially inserted in the core to provide effective control when criticality is achieved. The boron concentration in the Reactor Coolant System will then be reduced and criticality achieved by boron dilution or by subsequent rod withdrawal following boron dilution. Throughout this period, samples of the primary coolant will be obtained and analyzed for boron concentration.

Inverse count rate ratio monitoring, using data from the normal plant source range instrumentation, will be used as an indication of the proximity and rate of approach to criticality. Inverse count rate ratio data will be plotted as a function of rod bank position during rod motion and as a function of reactor makeup water addition during reactor coolant system boron concentration reduction.

14.2.10.3 Low Power Testing

Following initial criticality, a program of reactor physics measurements will be undertaken to verify that the basic static and kinetic characteristics of the core are as expected and that the values of the kinetic coefficients assumed in the safeguards analysis are conservative.

Procedures will specify the sequence of tests and measurements to be conducted and the conditions under which each is to be performed in order to ensure both safety of operation and the validity and consistency of the results obtained. If test results deviate significantly from design predictions, if unacceptable behavior is revealed, or if unexplained anomalies develop, the plant will be brought to a safe stable condition and the situation reviewed to determine the course of subsequent plant operation.

These measurements will be made at low power and primarily at or near normal operating temperature and pressure. Measurements will be made in order to verify the calculated values of control rod bank reactivity worths, the isothermal temperature coefficient, differential boron concentration reactivity worth, and critical boron concentrations. In addition, measurements of the relative power distributions will be made. For Unit 1, these measurements are conducted prior to exceeding 5% power. For Unit 2 the corresponding measurements are conducted prior to

exceeding 30% power. For Units 1 and 2, tests will be conducted on the instrumentation including power and intermediate range nuclear channels.

14.2.10.4 Power Level Escalation

After the operating characteristics of the reactor have been verified by low power testing, a program of power level escalation will bring the unit to its full rated power level in successive stages. The minimum test requirements for each successive stage of power escalation will be specified in the initial startup test procedures.

Measurements will be made to determine the relative power distribution in the core as functions of power level and control assembly bank position.

Secondary system heat balance measurements will ensure that the indications of power level are consistent and provide bases for calibration of the power range nuclear channels. The ability of the Reactor Coolant System to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations will be verified.

At prescribed power levels the dynamic response characteristics of the primary and secondary systems will be evaluated. System response characteristics will be measured for design step load changes, a rapid load reduction, and plant trips.

Adequacy of radiation shielding will be verified by gamma and neutron radiation surveys at selected points inside the Containment Building and throughout the station site at various power levels. Periodic sampling will be performed to verify the chemical and radio-chemical analysis of the reactor coolant.

14.2.11 TEST PROGRAM SCHEDULE

The sequential schedule for the preoperational testing of individual systems and components is shown in Figure 14.2-3. The sequential schedule for initial startup performed following fuel loading is shown in Figures 14.2-4A and B. These schedules show certain milestones at which time the tests, or portions of tests, will be completed and the overall time frame in which the tests will be conducted. The detailed schedules for testing will be prepared, reviewed and revised on a continuing basis as station construction progresses.

Preoperational testing of the various systems and components, which will continue up to fuel loading, is scheduled to commence from 10 to 18 months prior to fuel load. Initial startup tests are scheduled to be conducted over a period of approximately five months subsequent to fuel load. Fuel load, initial criticality, and power ascension tests are scheduled to be accomplished during this time period.

Station structures, systems, and components which are relied upon to prevent or mitigate consequences of postulated accidents will be fully tested to the extent practical prior to exceeding the 50 percent power level. Certain systems will have part of the preoperational testing performed after fuel load due to system configuration (e.g. control rod drive mechanism require fuel in the vessel to be fully tested). Such systems will be sufficiently tested prior to fuel load to provide reasonable assurance of successful testing after fuel load.

The development of test procedures will be an ongoing process, which will consist of preparation, review, and revision. The preparation of preoperational test procedures is scheduled to start no later than 15 months prior to fuel loading.

Preparation of initial startup procedures is scheduled to commence approximately five months prior to fuel loading. Preoperational test procedures will be available for examination by the NRC regional personnel approximately 60 days prior to the scheduled performance of the test. Initial startup test procedures will be available for examination by the NRC regional personnel approximately 60 days prior to the scheduled fuel loading date.

Testing efforts on either unit will not be diluted by activities on the other unit.

14.2.12 INDIVIDUAL TEST DESCRIPTIONS

14.2.12.1 Preoperational Tests

The test summaries for each of the preoperational tests to be performed, along with an index to these summaries, are provided in Table 14.2-2. These summaries describe the various tests which are specified as preoperational tests in Regulatory Guide 1.68 as discussed in Appendix 1A(B). The scope and titles of these summaries may not in all cases correspond directly to the actual test procedures which will be used during the Startup Program. Certain test procedures may include more than one test as described in these summaries, and in some cases tests described in one summary may be covered under more than one procedure. The overall scope and content of the tests described in these summaries will be addressed in final procedures.

There will be certain prerequisites which will apply in general to all preoperational tests. For convenience these general prerequisites are listed here rather than included in each summary.

General Prerequisites:

1. Construction activities have been completed on the system and on the necessary portions of supporting systems and any deficiencies have been properly dispositioned.

2. Functional operability of individual components or subsystems has been demonstrated for items such as valves, pumps, motors, instrumentation, and controls.

3. Test equipment necessary for test performance is available and calibrated.

4. Electrical power and air supplies are available as required for test performance.

14.2.12.2 Initial Startup Tests

The initial startup phase of the test program consists of preoperational tests deferred to the Initial Startup Program, Initial Startup Tests prior to and including fuel loading, precritical tests, initial criticality, low power physics tests, and power ascension tests. Fuel load and initial criticality procedures and tests are described in Section 14.2.10. Test summaries providing descriptions of all other major initial startup tests are provided in Table 14.2-3 along with an index for these test summaries.

TABLE 14.2-1
NOT USED

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TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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STATION SERVICE WATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of each train of the Station Service Water System to supply adequate cooling water flows to its required loads, including the CCW heat exchangers, for the subsystem associated with unit being tested.

PREREQUISITES

1. There is sufficient water in the Safe Shutdown Impoundment to provide adequate suction for the service water pumps.

TEST METHOD

1. Verify automatic starting of the standby SSW pump upon tripping of the operating SSW pump.

2. For Unit 1 only, verify proper operation of the recirculation line pressure controllers.

3. Check for proper functioning of instrumentation, alarms, and interlocks.

4. Confirm that the isolation valving systems will properly respond to an isolation signal.

5. Verify proper system flow balancing and SSW pump hydraulic performance.

ACCEPTANCE CRITERIA

The Station Service Water System Pumps meet design requirements. System flow and pressure requirements are satisfied. Instrumentation, controls, annunciators and interlocks function properly in response to simulated or normal input signals. The system meets design requirements when operating from the emergency power source.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 5 of 71)

DEMINERALIZED AND REACTOR MAKEUP WATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Demineralized and Reactor Makeup Water System to furnish deaerated demineralized water for use as reactor coolant.

PREREQUISITES

1. All necessary supporting equipment is operational.

TEST METHOD

1. Verify that the system is capable of producing demineralized water at the required rate.

2. Verify the proper operation of the vacuum deaerator and transfer pumps and confirm that the deaerated demineralized water is delivered to the Reactor Makeup Water Storage Tanks.

3. Verify proper operation of the reactor makeup water pumps and confirm their capability to supply reactor makeup water to all distribution points.

4. Demonstrate proper functioning of instrumentation, interlocks, and alarms.

ACCEPTANCE CRITERIA

The Demineralized Water System is capable of producing demineralized water which meets or exceeds the design criteria for water quality and capacity and is capable of supplying adequate quantities of demineralized, deaerated water to the system distribution points. Instrumentation, controls, annunciators and interlocks function properly in response to simulated or normal input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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COMPONENT COOLING WATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of each train of the Component Cooling Water System to supply adequate cooling water flows to its associated safety-related loads for the subsystem associated with unit being tested.

PREREQUISITES

1. Station Service Water System is operational

2. An acceptable water supply is available to component cooling water surge tanks.

TEST METHOD

1. Demonstrate all manual modes of operation including various pump/loop combinations.

2. Verify automatic starting of the standby CCW pump upon tripping of the operating CCW pump.

3. Demonstrate automatic isolation of the nonsafety-related loop from the rest of the subsystem, and isolation between the two safety related loops.

4. Check all other alarms and interlocks.

5. Verify proper system flow balancing of the safety and non- safety-related loops and CCW pump Hydraulic Performance.

6. Verify proper system flow balancing and CCW pump hydraulic performance.

ACCEPTANCE CRITERIA

The Component Cooling Water System pumps meet design requirements. System flow and pressure requirements are satisfied. The system supplies adequate cooling to both the safety and non-safety-related loops in the normal plant configuration and to the safety-related loops in the LOCA configuration. The system operates properly from the emergency power source. Instrumentation, controls, annunciators and interlocks, including system isolation functions, operate properly in response to normal or simulated input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 7 of 71)

PROCESS SAMPLING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Process Sampling System to provide liquid and gas samples through the correct flow path from all sample points in the primary and secondary systems, and to demonstrate the adequacy of sampling procedures and determine sample line holdup times.

PREREQUISITES

1. Plant conditions are established as necessary to facilitate drawing of liquid and gas samples from the required sampling locations.

2. Necessary tanks and sampling devices are available for receiving sample effluents and relief valve discharge.

TEST METHOD

1. Demonstrate proper system operation with regard to flow paths, flow capacity, and mechanical operability.

2. Verify the operability of the sample coolers and pressure reducing and regulating equipment.

3. Demonstrate the operability of the automatic on-line analyzers and in-line radiation monitors.

4. Check operation of instrumentation, interlocks, and alarms.

5. Demonstrate by flow rate vs line length that the sample line holdup time is within allowable limits.

6. Demonstrate adequacy of sampling procedures

ACCEPTANCE CRITERIA

The Secondary and Process Sampling System provides samples for analysis within the design requirements for flow rates, temperature and pressure. Sample line delay times are within acceptable limits. Analyzing equipment operates within design requirements. Instrumentation, controls, annunciators and interlocks function properly in response to simulated or normal input signals. The sampling procedures are adequate to provide proper samples and protect personnel.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 8 of 71)

VENTS AND DRAINS SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Vents and Drains System to collect and separate potentially radioactive and non-radioactive liquid wastes from valve and pump leakoffs, tank overflows, and tank drains.

PREREQUISITES

1. The Liquid Waste Processing System is capable of receiving liquid wastes and drainage.

TEST METHOD

1. Demonstrate the flow capability of the system.

2. Verify that all sump pumps are started and stopped automatically by the sump level controllers.

3. Check for proper operation of instrumentation, alarms, and interlocks.

ACCEPTANCE CRITERIA

The Vents and Drains System is capable of processing collected liquids at or exceeding design flow rates. Sump pump controls and interlocks function in accordance with design requirements. Potentially radioactive and non-radioactive liquids are separated as per the design criteria.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

FIRE PROTECTION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Fire Protection System, including its fire detection and fire suppression functions in accordance with design requirements.

PREREQUISITES

1. The Fire Protection System flow element and indicator and associated instrumentation is operational.

2. An acceptable treated water supply is available from the Fire Protection System Water Storage Tanks for fire pump and system testing.

TEST METHOD

1. Verify the proper functioning of the fire detection devices to activate the automatic fire protection system, alert the Control Room operators, initiate fire alarms, and to activate automatic closure of fire dampers, as required.

2. Verify proper operation of the fire suppression system and obtain flow rates through the underground loop.

3. Demonstrate the automatic start feature of the Fire Protection System pumps.

4. Verify proper flows of the Fire Protection System Pumps by use of the system flow element and indicator.

5. Demonstrate flow in wet pipe sprinkler systems.

6. Demonstrate proper operation of system instrumentation, alarms, and interlocks.

ACCEPTANCE CRITERIA

The Fire Detection portion of the Fire Protection System, in response to simulated input signals, provides indication, annunciation and/or fire suppression activation outputs in accordance with system design. The fire pumps meet or exceed the CPNPP Fire Protection Report requirements and provide flow to the water fire suppression system. The Halon Fire Suppression System interlocks, controls, annunciators, instrumentation and active fire isolation devices function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

Halon Fire Suppression Systems will be accepted utilizing either of the two methods below:

1. Testing and certification provided by the equipment supplier, and accepted by TU Electric or

2. Testing performed by TU Electric (or designated vendor), reviewed by JTG and approved by the Manager, Startup.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 11 of 71)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Spent Fuel Pool Cooling and Cleanup System to provide required cooling water flows to the spent fuel pool and verify proper operation of the purification loops and skimmers.

PREREQUISITES

1. As required for portions of the test, the Spent Fuel pool is filled with demineralized or borated water.

TEST METHOD

1. Verify proper operation and actuation of pumps and valves and verify correct flows in the spent fuel pool cooling loops.

2. Verify correct cooling water flows to the heat exchangers.

3. Verify proper operation and actuation of pumps and valves in the spent fuel pool cleanup loop and verify correct flows to filters and demineralizers.

4. Verify proper actuation and operation of the pump and skimmers in the spent fuel pool skimmer loop and verify correct flows in the loop.

5. Check operation of instrumentation, interlocks, and alarms.

6. Verify correct flows entering and leaving the system.

7. Verify proper operation of the spent fuel pool anti-siphon device.

ACCEPTANCE CRITERIA

The Spent Fuel Pool Cooling and Cleanup System pumps meet design requirements. System flow and pressure requirements are satisfied. Flows to heat exchangers, filters, demineralizers, and to and from the system are in accordance with design. The fuel pool anti-siphon devices and skimmer loop operates properly. Instrumentation, controls, annunciators and interlocks function properly in response to simulated or normal input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 12 of 71)

RESIDUAL HEAT REMOVAL SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Residual Heat Removal System and its capability to provide recirculation flows required to remove heat from the Reactor Coolant System.

PREREQUISITES

1. For portions of the test, as required, the RCS is filled with water.

TEST METHOD

1. Verify the Residual Heat Removal System (RHRS) inlet valve interlocks and trips operate properly.

2. Demonstrate the miniflow control valves operate properly.

3. Demonstrate acceptable pump performance while operating on the miniflow bypass line with the discharge isolation valves closed. Record sufficient data to insure pump operation is within acceptable regions of the pump manufacturer's flow curves.

4. Demonstrate recirculation capability within the isolated RHRS.

5. Demonstrate RHRS operation during heatup with letdown through RHRS.

6. Demonstrate RHRS operation during plant cooldown following hot functional testing.

7. Verify proper operation of instrumentation, interlocks and alarms.

Note: The safety injection functions of RHRS will be demonstrated during the testing of the Safety Injection System.

ACCEPTANCE CRITERIA

The Residual Heat Removal System and components function in accordance with design requirements. System flow capability in the various operating modes meets design requirements. Controls, interlocks, instrumentation and annunciators function properly in response to normal or simulated input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING, LETDOWN, AND SEAL WATER SUBSYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Chemical and Volume Control System, including capability to maintain charging and letdown flows, to maintain seal-water injection flow to the reactor coolant pumps, and to provide auxiliary spray flow to the pressurizer.

PREREQUISITES

1. Applicable portions of the Reactor Coolant System are capable of operationally interfacing with the Chemical and Volume Control System as required.

2. A cooling water supply is available for the heat exchangers as necessary for test performance.

3. Systems required to supply cover gas to the Volume Control Tank are operational and adequate supplies of gas are available.

TEST METHOD

1. Verify proper functioning of charging and letdown system components, including the hydraulic performance of the charging pumps, and operability of heat exchangers, letdown orifices, and control valves.

2. Demonstrate the capability to maintain seal water flow to the reactor coolant pumps, and the proper operation of seal water return and seal water bypass systems.

3. Verify proper flows and pressure drops for seal injection, seal water return, and reactor coolant filters.

4. Verify proper operation of excess letdown and alternate charging systems and components.

5. Verify proper operation of auxiliary spray to the pressurizer.

6. Verify proper operation of the volume control tank level control system, diversion valves, and cover gas system.

7. Check operation of instrumentation, interlocks, and alarms.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

ACCEPTANCE CRITERIA

The charging pumps meet design requirements. System flow and pressure requirements are satisfied. The charging and letdown normal and alternate flow paths, including heat exchangers, letdown orifices, and control valves, function in accordance with design requirements. The Volume Control Tank Level System, diversion valves and cover gas system function properly. Instrumentation, controls, annunciators and interlocks function properly in response to normal or simulated input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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CHEMICAL AND VOLUME CONTROL SYSTEM
CHEMICAL CONTROL, PURIFICATION, AND MAKEUP SUBSYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the subsystem of the Chemical and Volume Control System which is used to maintain control of water chemistry conditions, to remove gases from reactor coolant, to provide reactor makeup control, and to adjust reactor coolant boron concentration.

PREREQUISITES

1. Applicable portions of the Reactor Coolant System are capable of operationally interfacing with the CVCS as required.

2. Portions of systems supplying gas to the volume control tank are operational and adequate supplies of gas are available.

TEST METHOD

1. Demonstrate the chemical control function of the CVCS by verifying the capability of the system to introduce chemicals into the charging flow for pH and oxygen control, and that the system is capable of maintaining a gas pressure in the volume control tank as required during the applicable modes of operation.

2. Verify proper flows to mixed bed demineralizers, and determine pressure drops and effectiveness of demineralizers and filters.

3. Demonstrate the CVCS reactor make up control function, including maintaining a fluid inventory in the volume control tank, and testing of the automatic makeup, dilute, alternate dilute, borate, and manual modes of reactor makeup control.

4. Confirm proper functioning of pumps and control valves.

5. Check operation of instrumentation, interlocks, and alarms.

6. Verify proper operation of pumps, mixers and valves in the Boric Acid System.

7. Verify that a solution of boric acid can be mixed, transferred, recirculated and stored.

8. Verify that boric acid can be transferred to other systems as required.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

ACCEPTANCE CRITERIA

The chemical control, purification, and makeup subsystem, the boric acid subsystem, and components of the Chemical and Volume Control System operate in accordance with the design requirements. Reactor makeup control, gas pressure control and chemical control functions are demonstrated to meet design performance requirements. Instrumentation, controls, alarms and interlocks function properly in response to normal or simulated input signals. Boric acid is able to be batched, stored and transferred.

TABLE 14.2-2
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CHEMICAL AND VOLUME CONTROL SYSTEM
BORON THERMAL REGENERATION SUBSYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate the operability of the boron thermal regeneration subsystem.

PREREQUISITES

1. Applicable portions of the Reactor Coolant System are capable of operationally interfacing with the Chemical and Volume Control System.

2. A cooling water supply is available as required for heat exchangers.

TEST METHOD

1. Verify flows to and from thermal regeneration demineralizers, including proper operation of flow control valves, chiller units, heat exchangers, and filters.

2. Verify proper operation of water chillers and temperature control devices.

3. Check operation of instrumentation, interlocks, and alarms.

ACCEPTANCE CRITERIA

The boron thermal regeneration subsystem components of the Chemical and Volume Control System operates in accordance with design specifications. Flows to and from the thermal regeneration demineralizers and control of temperature into the demineralizers are in accordance with design requirements, instrumentation, controls, alarms and interlocks function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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BORON RECYCLE SYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate the operability of the Boron Recycle System.

PREREQUISITES

1. The recycle evaporator of the Boron Recycle System, and other interrelated or supporting equipment, are operational as required.

2. A steam supply and cooling water supply are available for the evaporator packages as required.

TEST METHOD

1. Demonstrate the capability of the Boron Recycle System to collect water from its various designated sources.

2. Verify proper operation of system components, including the evaporator package, the feed, distillate and concentrates pumps.

3. Verify the proper operation of the piping heat tracing.

4. Check for operability of instrumentation, alarms, and interlocks.

ACCEPTANCE CRITERIA

The Boron Recycle System components function in accordance with design specifications. Flow paths are verified operational. Heat tracing and other system related heating methods maintain system minimum temperatures per design requirements. Interlocks, alarms, controls and instrumentation function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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SAFETY INJECTION SYSTEM HYDRAULIC PERFORMANCE TEST SUMMARY

OBJECTIVE

Demonstrate the ability of the centrifugal charging pumps, the safety injection pumps, and RHR pumps to deliver required flows to the RCS during injection and recirculation modes of operation.

PREREQUISITES

1. The Refueling Water Storage Tank is filled with demineralized water or boric acid at refueling concentration.

2. The Reactor Coolant System is empty and the reactor vessel head is removed as required for full flow testing.

3. The Reactor Coolant System is closed and in a hot condition as required for portions of the test.

TEST METHOD - INJECTION MODE

1. Verify that all valves and components required for SIS operation are sequenced properly and are actuated within minimum required times. Verify proper operation of all instrumentation, interlocks and alarms.

2. With the RHR, centrifugal charging and SI pumps aligned to take suction from the RWST, sufficient data will be taken with the pump discharge valves shut to ensure satisfactory operation in the miniflow mode.

3. With reactor vessel head and internals removed, and the pumps taking suction from the RWST or the Reactor Coolant System hot legs (RHR pumps only), perform full flow tests for each pump with the miniflow valves closed. Measure flow and discharge pressure and adjust system as necessary to ensure that pumps do not exceed their maximum runout conditions. Measure flows in branch lines and adjust as necessary to ensure that flow distributions for all injection lines are within limits.

4. Align high head pumps to deliver to RCS hot legs. Measure and adjust flows as necessary.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

TEST METHOD - RECIRCULATION MODE

1. Ensure that suction lines from the Containment Building sumps are clean and free from obstruction.

2. Verify auto operation of RHR pump containment sump suction valves upon receipt of RWST lo-lo level signal.

3. Align safety injection and centrifugal charging pumps to take suction from the discharge of the RHR pumps and deliver to the RCS cold legs with RHR taking suction from the RWST or the Reactor Coolant System hot leg. Measure flows and adjust as necessary to ensure that RHR pumps deliver adequate flow to high head pump suction under runout conditions.

4. Verify RHR pump NPSH and anti-vortex capabilities using containment sump model testing.

ACCEPTANCE CRITERIA

The flow rates and flow distribution for all branch lines and the pump runout flows are within design limits as used in FSAR accident analyses with the discharge portion of the SIS aligned for injection and recirculation modes of operation. Instrumentation, controls, interlocks and alarms function properly in response to normal or simulated input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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SAFETY INJECTION ACCUMULATORS TEST SUMMARY

OBJECTIVE

Verify the discharge characteristics and proper system actuation for each of the SI accumulators.

PREREQUISITES

1. The Reactor Coolant System is drained down and the reactor vessel head and internals are removed for the full flow portion of the test, and is closed and in a hot condition as required for portions of the test relating to check valve hot operability.

2. A supply of nitrogen or compressed air is available for pressurizing the accumulators.

3. The accumulators are filled to their normal operating level with demineralized water or boric acid to refueling concentration.

TEST METHOD

1. Each accumulator is partially pressurized and its discharge valve is then opened allowing discharge into the RCS. Level and pressure measurements are used to calculate line resistance and accumulator injection performance.

2. Verify operability of nitrogen fill, venting and relief valves, accumulator drains, and accumulators makeup systems.

3. Verify ability of the accumulator isolation valves to open automatically on a safety injection signal against a differential pressure equal to the accumulator at maximum operating pressure and the Reactor Coolant System depressurized.

4. Verify operability of check valves in the accumulator discharge lines with the RCS in a hot condition.

5. Verify proper operation of all instrumentation, interlocks, and alarms.

ACCEPTANCE CRITERIA

Accumulator discharge performance is within design limits used for FSAR accident analyses. The accumulator systems function in accordance with design requirements. Controls, interlocks, alarms and instrumentation function properly in response to simulated or normal input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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CONTAINMENT SPRAY SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the hydraulic performance of the containment spray pumps, proper functioning of spray nozzles and headers, and the proper operation and actuation of the system components including the chemical additive system.

PREREQUISITES

1. A water supply is available for the containment spray pumps.

2. An air supply is available for testing the containment spray nozzles.

3. The containment spray pumps discharge lines are isolated to prevent spraying water into the Containment Building.

4. Verify that the piping between valves ICT-4776 and manual isolation valves ICT-141; and IHV-4777 and manual isolation valve ICT-146 has been satisfactorily flushed.

TEST METHOD

1. Through test connections, pass air under pressure to the spray nozzles and ring headers to ensure they are free of obstructions.

2. Operate the containment spray pumps through the RWST recirculation path to verify pump hydraulic performance.

3. Verify proper sequencing of valves and pumps upon safeguards actuation signals.

4. Demonstrate that the containment spray chemical additive system will automatically actuate and function as required.

5. Verify proper performance of chemical eductors and chemical additive tank vacuum breakers.

ACCEPTANCE CRITERIA

Containment spray pumps meet hydraulic design criteria. Unobstructed flow paths are verified for the containment spray nozzles and ring headers. Sequencing of all valves and pumps upon safeguards action signals occurs properly. Chemical additive system functions in accordance with design requirements. Controls, interlocks, instrumentation and annunciators, in response to normal or simulated signals, function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 23 of 71)

INTEGRATED ENGINEERED SAFETY FEATURES TEST SUMMARY

OBJECTIVE

Demonstrate proper automatic alignment and operation of all safeguards systems upon a safety injection signal using both normal offsite and emergency power sources.

PREREQUISITES

1. The RCS is cold and drained down and the reactor vessel head and internals are removed.

2. The Refueling Water Storage tank has an adequate supply of demineralized water or boric acid at refueling concentration.

3. The containment spray pump discharge lines are manually isolated to prevent spraying into the Containment Building.

4. Actuation circuitry has been tested and is capable of actuating all equipment upon a manually initiated safety injection signal.

5. All systems required to actuate on a safety injection signal are operational and are aligned for normal operation.

TEST METHOD

1. With the SIS aligned for normal power operation, manually initiate a safety injection signal ("S" signal) and verify proper starting of all safeguards pumps and measure flowrates. Verify proper alignment of valves, including centrifugal charging pump suction and discharge valves, and pump mini-flow lines.

2. Verify proper actuation, alignment, and operation of all systems which are required to operate or change mode of operation upon receipt of a safety injection signal including the Station Service Water System, Component Cooling Water System, and required ventilation systems.

3. Repeat test with no offsite power supply available and verify proper operation of emergency diesel generators in conjunction with safety injection actuation including proper load sequencing, pump start times, and valve sequencing.

ACCEPTANCE CRITERIA

The safeguards systems actuate in the proper sequence, automatically align in the proper manner, and provide required flowrates to the RCS injection points in accordance with design requirements.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

LIQUID WASTE PROCESSING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Liquid Waste Processing System to collect and process potentially radioactive wastes for recycle or for release to the environment.

PREREQUISITES

1. All necessary supporting equipment is operational.

TEST METHOD

1. Demonstrate capability of the Liquid Waste Processing System to process waste water in accordance with design requirements.

2. Demonstrate proper operation of all components in the system, including the waste evaporator, waste holdup tanks, system demineralizers, and laundry reverse osmosis system.

3. Verify the proper operation of the reactor coolant drain tank and associated equipment.

4. Check for proper functioning of instrumentation, interlocks, controls, and alarms.

ACCEPTANCE CRITERIA

The Liquid Waste Processing System collects, segregates and processes liquid wastes in accordance with design requirements. Operation of system components meet design specifications. Instrumentation, controls, annunciators and interlocks function properly in response to simulated or normal input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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SOLID WASTE PROCESSING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the subsystem used to transfer spent resins, evaporator concentrates, and chemical drain tank effluents and of the baling subsystem used to package low-radiation level compressible wastes.

PREREQUISITES

1. All necessary supporting equipment is operational.

2. Equipment, such as pumps, piping, baling units and associated controls and interlocks are operational.

TEST METHOD

1. Demonstrate the operability of the waste baling units.

2. Demonstrate the ability to transfer spent resins evaporator bottoms and chemical drain tank effluents from their source to the bulk disposal outlet.

3. Verify proper operation of associated instrumentation and interlocks.

ACCEPTANCE CRITERIA

The Solid Waste System is capable of transferring waste forms to the bulk disposal outlet and baling low level compressible waste. Controls, interlocks, and instrumentation function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 26 of 71)

GASEOUS WASTE PROCESSING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Gaseous Waste Processing System including its capability to remove and process gases from specified sources, including the volume control tank, boron recycle evaporator, reactor coolant drain tank, and waste evaporator.

PREREQUISITES

1. All necessary supporting equipment is operational.

TEST METHOD

1. Demonstrate proper operation of the system and its components, including filling and isolation of the gas decay tanks, waste gas compressor performance, and the catalytic recombiner operation.

2. Check for proper functioning of instrumentation, interlocks, and alarms.

ACCEPTANCE CRITERIA

The Gaseous Waste Processing System components meet design operational requirements. Flow paths and flow rates are in accordance with design criteria. Instrumentation, interlocks, controls and alarms function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 27 of 71)

CONTAINMENT ATMOSPHERE MONITORING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Containment Atmosphere Monitoring System.

PREREQUISITES

1. Setpoints for alarms and indicating lights have been set.

TEST METHOD

1. Demonstrate proper operation of the hydrogen analyzer/monitoring equipment by functional testing from the microprocessor keyboard.

2. Demonstrate proper operation of the temperature and humidity instrumentation.

3. Demonstrate proper operation of the containment pressure transmitters.

ACCEPTANCE CRITERIA

The Containment Atmosphere Monitoring System functions in accordance with design requirements.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

PROCESS RADIATION MONITORING SYSTEM TEST SUMMARY
(UNIT 1)

OBJECTIVE

To demonstrate the capability of the Process Radiation Monitoring System to continuously monitor radiation levels attributed to plant processes and plant effluents, record radiation levels, and initiate radiation alarm signals.

PREREQUISITES

1. A remotely operable check source is available for each radiation detector as required.

TEST METHOD

1. Verify proper operation and response of the radiation monitors.

2. Check for proper setting of the low-level radiation alarm setpoints.

3. Verify proper operation of Control Room high radiation level alarms.

4. Ensure proper operation of indicators and recording devices.

ACCEPTANCE CRITERIA

The radiation monitors respond properly to radiation check sources. Alarm setpoints are in accordance with design criteria. Remote instrumentation and recording devices operate in accordance with design requirements.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

PROCESS RADIATION MONITORING SYSTEM TEST SUMMARY
(UNIT 2)

OBJECTIVE

To demonstrate the capability of the Process Radiation Monitoring System to continuously monitor radiation levels attributed to plant processes and plant effluents, record radiation levels, and initiate radiation alarm signals.

PREREQUISITES

1. A remotely operable check source is available for each radiation detector as required.

2. Each radiation monitor and detector(s) has been calibrated using a radioactive source of known activity.

TEST METHOD

1. Verify proper operation and response of the radiation monitors.

2. Check for proper setting of the low-level radiation alarm setpoints.

3. Verify proper operation of Control Room high radiation level alarms.

4. Ensure proper operation of indicators and recording devices.

ACCEPTANCE CRITERIA

The radiation monitors successfully pass their calibration tests. Alarm setpoints are in accordance with design criteria. Remote instrumentation and recording devices operate in accordance with design requirements.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

AREA RADIATION MONITORING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Area Radiation Monitoring System to provide continuous surveillance of radiation dose rates throughout accessible areas of the plant, alert personnel of excessive dose rate levels, and provide direct reading indication and recording of dose rates at each monitor location.

PREREQUISITES

1. Setpoints for radiation alarms and indicating lights have been set.

2. A remotely operable check source is available for each radiation detector as required.

TEST METHOD

1. Verify proper operation of the Area Radiation Monitoring System, including the indicating and recording functions of the system.

2. Check for proper setting of the radiation alarm setpoints.

3. Verify that the Control Room alarms will annunciate both high radiation levels and circuit failures.

4. Ensure proper operation of instrumentation, alarms, and interlocks.

ACCEPTANCE CRITERIA

Radiation monitors respond properly to radiation check sources. Alarm setpoints are in accordance with design criteria. Remote instrumentation, recording devices and interlocks operate in accordance with design requirements.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

CONTROL ROOM VENTILATION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Control Room Ventilation System to maintain suitable and safe ambient conditions for operating equipment and personnel in the Control Room and associated areas.

PREREQUISITES

1. A cooling water supply is available for the Control Room condensing air compressor units.

2. Compressed air is available for system valve and damper operators.

TEST METHOD

1. Demonstrate proper functioning of the Control Room Ventilation System in the normal emergency recirculation and emergency ventilation modes of operation.

2. Verify that the system responds properly to safeguards actuation signals.

3. Verify proper pressure differentials.

4. Demonstrate proper operation of chlorine gas detectors, fans, valves, dampers, and filters.

5. Check operation of system instrumentation, interlocks, and alarms.

ACCEPTANCE CRITERIA

The Control Room Ventilation System operates in accordance with design requirements. Flow and pressure requirements are met. Filtering systems exceed requirements. Alarms, detectors, and interlocks function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

AUXILIARY, FUEL, AND SAFEGUARDS BUILDING VENTILATION TEST SUMMARY

OBJECTIVE

To demonstrate the proper operation of the ventilation systems for the Auxiliary Building, Safeguards Building, and Fuel Building.

PREREQUISITES

1. A cooling water supply is available for the cooling coils.

2. Compressed air is available for valve damper operators.

TEST METHOD

1. For the Auxiliary Building, Safeguards Building, and Fuel Building ventilation systems, demonstrate proper operation of the ventilation fans and filters, and verify adequate air flow to areas served by each system.

2. Demonstrate proper operation of the inlet and exhaust isolation dampers associated with equipment rooms in the Safeguards Building.

3. Demonstrate proper operation of the auxiliary cooling units in the Safeguards Building equipment rooms.

4. Verify the system’s ability to maintain proper building pressure differentials.

5. Ensure proper operation of instrumentation, interlocks, and alarms.

6. Demonstrate adequate performance of each ESF pump room cooler.

ACCEPTANCE CRITERIA

The Auxiliary Building, Safeguards Building, and Fuel Building ventilation systems operate in accordance design requirements. Air flows and pressure differentials are within design specification. The isolation dampers, alarms and interlocks function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 33 of 71)

CONTAINMENT VENTILATION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Containment Ventilation System to provide containment air-recirculation, control rod drive mechanism cooling, neutron detector well and nozzle support cooling, preaccess filtration, containment purging, hydrogen purging, and reactor coolant pipe penetration cooling.

PREREQUISITES

1. A cooling water supply is available for the fan-cooling units of the system.

2. For testing portions of the system as applicable, the control rod drive mechanisms and neutron detector are capable of being energized, and plant conditions are established as required.

3. Compressed air is available for valve and damper operators.

TEST METHOD

1. Demonstrate the capability of the Containment Ventilation System to provide for containment recirculation and heat removal, by testing operation of the axial fans, centrifugal water chillers and the cooling coils, and by ensuring adequate flow is delivered to components and areas inside Containment as required.

2. Verify that the control rod drive mechanism shroud ventilation units are capable of maintaining temperatures within the CRDM shroud within design limits.

3. Demonstrate proper operation of the fans, cooling coils, and inlet and outlet dampers associated with the neutron detector well and nozzle support cooling system, including verification during Hot Functional Testing and that the cooling system will maintain neutron detector temperature within specifications.

4. Confirm the capability of the containment preaccess filtration units to provide air circulation and filtration throughout all containment areas as required.

5. Verify proper operation of the containment purge supply and purge exhaust equipment, including functioning of the heating coils and fans.

6. Demonstrate proper operation of all Containment Ventilation System instrumentation, interlocks and alarms.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

7. Demonstrate during hot functional testing that the reactor coolant primary shield pipe penetration cooling system maintains concrete within design temperature limits.

8. Verify proper operation of the Hydrogen Purge Supply and Purge Exhaust systems.

ACCEPTANCE CRITERIA

The Containment Ventilation System components function in accordance with design requirements. Adequate ventilation flow is provided to containment areas to maintain or limit temperatures to design valves. System interlocks, instrumentation, alarms and controls operate properly. The reactor coolant pipe penetration concrete shall be maintained less than design limits as specified in FSAR Section 9.4A.2.7.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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COMBUSTIBLE GAS CONTROL SYSTEMS TEST SUMMARY

OBJECTIVE

To demonstrate the proper operation of the Hydrogen Recombiners and the Hydrogen Purge System.

PREREQUISITES

1. The hydrogen recombiners are operable and their associated power panel is energized.

2. The Hydrogen Purge System is operational.

TEST METHOD

1. Verify the proper operation of the hydrogen recombiners, including the heaters, controllers and temperature indicators.

2. Verify proper operation of the Hydrogen Purge System, including the system fans and filters.

ACCEPTANCE CRITERIA

The combustible gas control systems operate in accordance with design criteria. The hydrogen recombiners, system fans and filters perform their design function. Instrumentation, controls, annunciators and interlocks function properly in response to normal or simulated signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

ELECTRICAL AREA AND BATTERY ROOM VENTILATION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the proper operation of the Electrical Area and Battery Room Ventilation Systems.

PREREQUISITES

1. Supporting systems are operational as required.

2. Compressed air is available for valve and damper operations.

TEST METHOD

1. For the Electrical Area and Battery Room Ventilation System, demonstrate proper operation of the ventilation supply and exhaust fans, heating and cooling coils, shutoff louvers and filters, and adequate flow to areas served by each portion of the system.

2. Demonstrate operability of the exhaust fans and ducts with each battery room.

3. Verify proper operation of the Uncontrolled Access Area Ventilation System and demonstrate its capability to supply ventilation airflow to each of the battery rooms.

ACCEPTANCE CRITERIA

The Electrical Area and Battery Room Ventilation Systems operate in accordance with design requirements and provide adequate ventilation flows. Components which receive emergency power operate properly when supplied from their emergency power sources.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

DIESEL GENERATOR COMPARTMENT VENTILATION SYSTEMS TEST SUMMARY

OBJECTIVE

To demonstrate the capability of each diesel generator compartment ventilation system to provide adequate ventilation for its associated diesel generator compartment.

PREREQUISITES

1. All necessary supporting equipment is operational.

TEST METHOD

1. Demonstrate proper operation of the system intake and exhaust louvers and exhaust fans.

2. Verify that the ventilation system exhaust fans start automatically on receipt of a diesel generator start signal.

3. Verify adequate ventilation during Emergency Diesel Generator operations.

4. Check operation of instrumentation and alarms.

ACCEPTANCE CRITERIA

The Diesel Generator Compartment Ventilation Systems operate in accordance with design requirements and maintain the diesel generator compartment space temperature at or below design maximum when the diesel generators are operating. Instrumentation, controls, annunciators and interlocks function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

DIESEL GENERATORS TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the diesel generator unit and associated auxiliaries, including the proper starting and load sequencing.

PREREQUISITES

1. DC power is available for controls, alarms, protective relays, air starting solenoid valves and generator field flashing.

2. Compressed air is available for air starting of the diesel generator unit.

TEST METHOD

1. Verify proper performance of the diesel generator fuel oil storage and transfer system, cooling and heating systems, and the lubricating oil system.

2. Demonstrate the capability of the unit to start manually and automatically as designed and attain the required voltage and frequency within the acceptable time limitation.

3. Verify the capability of the diesel generator unit to accept the sequenced equipment by utilizing actual plant loads appropriate for existing plant conditions. The loads will be added at the proper sequence and time duration, while maintaining an acceptable voltage and frequency.

4. Verify proper operation of voltage and frequency controllers. Perform load rejection tests as required and verify proper operation of overspeed control devices.

5. Perform at least 35 valid start and load tests per diesel generator with no failures from cold ambient conditions to at least 50 percent continuous rating.

6. Demonstrate loading capability for at least 24 hours, of which 22 hours will be a load equivalent to the diesel generator continuous rating and 2 hours will be a load equivalent to the diesel generator 2-hour rating.

7. Demonstrate the ability to synchronize the diesel generator unit with offsite power while the unit is connected to the emergency load, transfer this load to the offsite power, isolate the diesel generator unit, and restore it to standby status.

8. Demonstrate that the diesel generator capability to supply emergency load is not impaired during periodic testing.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

9. Demonstrate functional capability at full-load conditions by reperforming the automatic start and load sections of Steps 2 and 3 immediately after Step 6 is completed.

10. Demonstrate that no common failure modes exist by starting the redundant diesel generators simultaneously.

ACCEPTANCE CRITERIA

The diesel generators supply emergency power to the 6.9KV Class 1E buses on loss of the preferred and alternate offsite power feeds to the emergency buses. The diesels start, attain speed and voltage and accept emergency load or rated load as designed. The fuel oil, lubricating oil, cooling and heating systems function as designed. Interlocks, alarms, controls and tripping devices function properly in response to normal or simulated input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

AC POWER DISTRIBUTION TEST SUMMARY

OBJECTIVE

To demonstrate that the AC Power Distribution System provides a reliable power source to safety-related loads, and that no interaction occurs between redundant trains.

PREREQUISITES

1. Control power is available for necessary switchgear.

2. Diesel generator sets and supporting auxiliaries, including DC power, are operational, where applicable.

3. Metering and relaying circuits are calibrated.

TEST METHOD

1. Confirm that loss of either redundant safety-related train will not impair the ability of the remaining system to supply power to required safety-related loads.

2. Verify performance of alarms, tripping devices, and interlocks.

3. Check local and remote operation and indication of switchgear.

4. Verify proper operation of manual/automatic transfer schemes for;

    a. For Class 1E 6.9KV Buses -

        1. Automatic transfer, slow, from preferred off-site source to alternate off-site source. Automatic transfer, slow, from preferred off-site to on-site, and from alternate off-site to on-site sources.

        2. Manual transfer, live, from any one of the three sources - preferred off-site, alternate off-site and on-site - to the other (six combinations).

    b. For Non-Class 1E 6.9kV Buses

        1. Automatic transfer fast and slow from unit auxiliary transformer to off-site source.

        2. Manual transfer from off-site source to unit auxiliary transformer and vice versa.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

5. Verify proper load shedding and/or undervoltage tripping of all 6.9kV and 480V switchgear.

6. Verify load-carrying capability of Class 1E and non-Class 1E transformers, cables, switchgear and feeder breakers.

ACCEPTANCE CRITERIA

The AC Power Distribution System provides power to safety-related loads when supplied from normal and emergency power sources. No interaction occurs between redundant trains. Automatic transfer of power supplies to vital buses occurs properly and meets design requirements. Interlocks, alarms, controls and tripping devices function properly in response to normal or simulated input signals. Emergency bus voltages are maintained within design limits under load.

TABLE 14.2-2
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DC POWER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the ability of the two redundant 125 VDC systems to supply power to safety-related loads during normal plant conditions and upon loss of all AC power system.

PREREQUISITES

1. Batteries are maintained in a nominally fully charged condition.

2. AC power is available to battery chargers.

3. Protective devices (including ground detection), breakers, and system status sensing devices, relays and meters have been calibrated for desired response to abnormal system conditions.

4. Distribution components, breakers, cables and panels have been tested and verified energized, free of grounds, providing nominal DC power to all Class 1E DC loads.

TEST METHOD

1. Confirm that the batteries have sufficient capacity to carry essential loads continuously without battery chargers for the minimum required time, and that individual cell limits are not exceeded.

2. Confirm that loss of either redundant safety-related train will not impair the ability of the remaining system to supply power to safety-related loads.

3. Check functioning of alarms, local and remote indicators, breaker status indications, and other protective devices including transfer devices and interlocks.

4. Demonstrate the ability of battery chargers to recharge the batteries from the design minimum charge state to the fully charged state within 24 hours while supplying the steady state loads under all modes of plant operation.

ACCEPTANCE CRITERIA

The batteries demonstrate the ability to carry the essential loads continuously for the design time period with the battery chargers out of service. The battery chargers maintain the batteries at full charge and are demonstrated to be able to restore the batteries to full charge within design time limits. Protective devices, controls, interlocks, alarms and instrumentation function properly in response to normal or simulated inputs.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

INSTRUMENTATION AND CONTROL POWER SUPPLY SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Class 1E I&C Power Supply System, including the proper functioning of the system inverters.

PREREQUISITES

1. 125 VDC power supplies are available to the inverter units.

2. The common backup source of 120 VAC power is available to the instrument distribution panels.

TEST METHOD

1. Verify that the system inverters maintain the current and voltage to the instrument distribution panels within the required limits.

2. Demonstrate that the loss of either redundant I&C Power Supply System or its associated power supplies will not cause a loss of power to more than one of the redundant vital buses.

3. Check operation of all interlocks and alarms.

4. Upon loss of the normal power supply, demonstrate the capability of the inverters to maintain output by means of the alternate power supply.

ACCEPTANCE CRITERIA

The Instrumentation and Control Power Supply System operates in accordance with design requirements. Inverters transfer to alternate power supplies at the correct setpoint. Redundant trains are verified independent. Alarms and interlocks function properly.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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COMPUTER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate that both the hardware and software present accurate indications of plant parameters and processed information.

PREREQUISITES

1. The computer system has been installed as necessary for test performance

2. Inputs or simulated inputs are available for testing

TEST METHOD

1. Verify inputs are connected correctly

2. Verify that both hardware and software processing of input signals results in accurate information

ACCEPTANCE CRITERIA

The computer system records and displays accurate information from plant parameters using both direct inputs and processed information.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

DC EMERGENCY LIGHTING SYSTEMS TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the DC Emergency Lighting System to provide lighting in those locations where safety-related functions are performed.

PREREQUISITES

1. DC power is available to the Control Room DC Emergency Lighting System.

2. The battery-pack operated lighting units are in service.

TEST METHOD

1. Demonstrate proper operation of the DC Emergency Lighting System in the Control Room.

2. Verify the operability of the 8 hour battery-pack lighting units and integral automatic charger.

ACCEPTANCE CRITERIA

The DC Emergency Lighting System operates in accordance with design requirements, supplying illumination to the required areas. The DC powered lighting initiates when AC powered emergency lighting is lost.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

COMMUNICATION SYSTEMS TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the intraplant and plant-to-offsite communication systems to provide adequate communication coverage and audibility.

PREREQUISITES

1. As required to check the audibility of intraplant communication systems, equipment is operating to create an ambient noise level that would be expected during normal plant operation.

TEST METHOD

1. Demonstrate proper functioning of the Public Address System, including the capability of the voice paging channel output to be clearly audible over the highest expected noise levels.

2. Verify proper operation of the Intraplant Telephone System and Intraplant Sound-Powered Telephone System.

3. Demonstrate operation of the Offsite Communication Systems, including the public telephone and two-way radio systems and the direct telephone line to the system dispatcher.

4. Verify proper functioning of the Emergency Evacuation Alarm System.

ACCEPTANCE CRITERIA

The communication systems function in accordance with design requirements. The Public Address and Emergency Evacuation Alarm Systems are audible during the highest expected ambient noise levels. Intraplant communications channels function properly and communications to offsite can be established by public telephone, two- way radio and direct telephone line to the system dispatcher.

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REACTOR CONTROL SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Reactor Control System to respond properly to simulated input signals and to transmit proper control signals to other plant control systems and components.

PREREQUISITES

The Reactor Control System is energized, calibrated, and aligned in accordance with the test documents.

TEST METHOD

1. Demonstrate proper operation of the Reactor Control System under various simulated plant conditions.

2. Perform tests utilizing signals or simulated signals on each of the Reactor Control System inputs in accordance with the applicable manufacturer's instruction manual.

3. Check each individual control channel from its input to its output signal to other plant control systems and components.

   NOTE: A functional demonstration at approximately 50% power is performed to verify that the Reactor Coolant System automatically maintains the proper reactor coolant average temperature conditions (see Table 14.2-3)

ACCEPTANCE CRITERIA

The Reactor Control System responds properly to simulated or normal input signals to interfacing systems. Rod blocks and permissives are verified correct in accordance with design.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 48 of 71)

REACTOR PROTECTION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Reactor Protection System and Engineered Safety Features Actuation System to respond properly to logic initiation signals prior to initial fuel loading.

PREREQUISITE

1. The instrument and protection systems are energized, calibrated, and aligned.

TEST METHOD

1. Demonstrate proper operation of the Reactor Protection System and Engineered Safety Features Actuation System under various logic conditions.

2. Determine the process fluid variable to sensor input delay time analytically.

3. Perform tests utilizing signals or simulated signals on each of the Reactor Protection System analog inputs in accordance with the applicable specifications.

4. Verify response time of the logic channels. Signals will be input to the primary sensor and timed through to the Reactor Trip Breaker.

5. Demonstrate redundancy and safe failure on loss of power.

6. Verify coincidence for all logic conditions.

ACCEPTANCE CRITERIA

The Reactor Protection System and Engineered Safety Features Actuation System function in accordance with design specifications in response to logic initiation signals. Total response time of the logic channels including process fluid delay time is in accordance with technical specification requirements.

TABLE 14.2-2
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EXCORE NUCLEAR INSTRUMENTATION TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Excore Nuclear Instrumentation including its ability to supply signals to the Reactor Protection System for generating appropriate trip signals and alarms, and indicating reactor power levels.

PREREQUISITES

1. Prior to Fuel Load.

TEST METHOD

1. Demonstrate the capability of source, intermediate, and power range circuitry to respond to a simulated test signal.

2. Verify that the source range detectors properly respond to a neutron source.

3. Demonstrate proper operation of the auctioneering circuits and flux channel deviation signals.

4. Check all channels to verify high level trip functions, alarm setpoints, and audible count rates where applicable.

ACCEPTANCE CRITERIA

The Excore Nuclear Instrumentation responds with design specifications to simulated input signals and provides proper output to the Reactor Protection System. Trip and alarm set points are in accordance with predetermined limits and proper neutron flux levels are displayed. The source range responds properly to a neutron source.

NOTE: Refer also to Section 14.2.10.

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PREOPERATIONAL TESTS SUMMARIES INDEX

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SATURATION MARGIN MONITOR TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Saturation Margin Monitor to accurately calculate and display Saturation Margin and to display hottest sensor temperature. Additionally, to demonstrate accuracy of the in-core thermocouples.

PREREQUISITES

1. Plant conditions are established as required.

2. Reactor Coolant System resistance temperature detectors are operational.

3. RCS Hot Leg and pressurizer pressure channels are operational.

4. The plant/ERF computers are operational.

TEST METHOD

1. Verify the identification of each thermocouple.

2. Measure the resistance and verify the continuity of each thermocouple.

3. Calibrate the incore thermocouples to the average of the RCS resistance temperature detectors.

4. Check the incore thermocouples temperature compensation equipment.

5. Verify that incore thermocouple signals are properly transmitted to the plant/ERF computers.

6. Verify that the Core Cooling Monitor responds properly to simulated input signals and that Saturation Margin and hottest sensor temperature are accurately calculated and displayed on the control board and computers.

ACCEPTANCE CRITERIA

The incore thermocouples achieve satisfactory calibration and provide proper inputs for the core cooling monitor and computers. Saturation margin and hottest sensor temperature are accurately calculated and displayed on the control board and computers. Core exit temperatures are accurately displayed on the computers.

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PREOPERATIONAL TESTS SUMMARIES INDEX

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AUXILIARY STARTUP INSTRUMENTATION TEST SUMMARY

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PREOPERATIONAL TESTS SUMMARIES INDEX

SEISMIC INSTRUMENTATION TEST SUMMARY

OBJECTIVE

To demonstrate the operability of installed seismic instrumentation and its capability to monitor and record seismic disturbance.

PREREQUISITES

1. A calibrated seismic test signal is available.

TEST METHOD

1. Align the seismic instrumentation to a calibrated test signal.

2. Verify that the seismic trigger will activate the instrumentation as required at an acceptable level.

3. Demonstrate proper operation of seismic instrumentation components and alarms, including the triggering device, recording and playback system, peak acceleration recorders, and seismoscopes.

ACCEPTANCE CRITERIA

The seismic instrumentation is capable of being aligned within specification requirements. The system instruments, including trip settings, demonstrate correct response and outputs in response to simulated input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

STEAM GENERATOR SAFETY & RELIEF VALVES TEST SUMMARY

OBJECTIVE

To demonstrate the proper functioning of the safety valves and power-operated relief valve associated with each steam generator.

PREREQUISITES

1. Plant conditions are established as necessary for test performance.

2. A pressure-assist device is available for use in lifting the relief and safety valves, and equipment is available to measure lifting and reseating pressures and valve stroke time.

TEST METHOD

1. Verify proper actuation and operation of the power-operated relief valves at normal system operating pressure by simulating a pressure signal to the controller.

2. With steam pressure near the safety valve setpoint, lift each safety valve with a pressure-assist device and measure the lifting pressure to ensure setpoints are as required.

3. Determine the full stroke time of the power-operated relief valves at normal system operating pressure by simulating a pressure signal to the controller.

4. With the steam generators at normal operating pressure, verify that safety valve leakage is within acceptable limits, the valves seat properly and do not chatter by visual and audio observation.

ACCEPTANCE CRITERIA

The steam generator safety valves and power-operated relief valves operate in accordance with design requirements. Safety valve lift settings meet technical specification requirements. Valves reset properly and do not chatter. Safety valve leakage at normal operating pressure is within design criteria.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

MAIN STEAM AND FEEDWATER ISOLATION VALVES TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the main steam and feedwater isolation valves including their capability to close automatically as required.

PREREQUISITES

1. Plant conditions are established as necessary for test performance.

2. Equipment is available to measure the closure time of the isolation valves.

TEST METHOD

1. Demonstrate remote shutdown panel and Control Room operation of the main steam isolation valves and Control Room operation of the feedwater isolation valves.

2. Verify that the isolation valves will close upon receipt of an isolation signal.

3. Measure the closure time of the main feedwater isolation valves to ensure they close within the maximum and minimum time required and measure the closure time of the main steam isolation valves to ensure they close within the maximum allowed time.

4. Verify that the main steam isolation valves will close upon receipt of an isolation signal after having been isolated from the Instrument Air System for a minimum of 30 minutes.

ACCEPTANCE CRITERIA

The main steam and feedwater isolation valves respond properly to remote shutdown panel and control room operation, as applicable, and close upon receipt of an isolation signal. Valve closure times meet Technical Specification requirements.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

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AUXILIARY FEEDWATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability and reliability of the Auxiliary Feedwater System to supply feedwater to the steam generators and to maintain steam generator water inventory as required.

PREREQUISITES

1. Steam supply is available for the turbine-driven Auxiliary Feedwater pump.

2. An acceptable water supply is available for the Condensate Storage Tank.

TEST METHOD

1. Demonstrate proper manual and automatic operation of suction and discharge valves including steam supply to the turbine-driven pump.

2. Verify proper functioning of local and remote means of component control.

3. Verify that the hydraulic performance of each pump meets design requirements.

4. Demonstrate that the motor driven and turbine-driven pumps are capable of delivering flow to the steam generators within the acceptable time after an initiating signal.

5. Check for proper operation of instrumentation, control interlocks and alarms.

6. Verify the capability of the flow limiter to prevent pump runout.

7. Verify the capability of the Station Service Water System to supply water to the Auxiliary Feedwater Suction Header.

8. Verify that each Auxiliary Feedwater Control Valve can isolate the Auxiliary Feedwater to the Steam Generators for a minimum of 30 minutes while isolated from the Instrument Air System.

9. Verify system reliability with five consecutive, cold quick starts.

10. Demonstrate the ability of the AFWS to restore steam generator level after a low-level transient without causing unacceptable feedwater/steam generator waterhammer.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

11. Perform a 48 hour endurance on each of the auxiliary feedwater pumps after achieving the following conditions:

    a. Pump achieves operating speed, rated discharge pressure, and flow. A reduced flow not less than vendor limitations is acceptable.

    b. Steam temperature for the turbine driven AFW pump to be no less than 400°F. The AFW pump turbine is to be operated for a minimum of two hours without forced ventilation.

    c. Upon completion of 48 hour run the pump shall be shut down and allowed to cool to within 20°F of initial conditions, but not less than eight hours. The pump shall then be started and allowed to run for one hour.

ACCEPTANCE CRITERIA

The Auxiliary Feedwater System operates in accordance with technical specification requirements, and its performance is within limits assumed in the appropriate accident analysis. System automatic controls function properly in response to normal or simulated signals. Controls, interlocks, annunciators and instrumentation function in accordance with design requirements. Flow limiters limit runout flow to design values. The control valves function for a minimum of 30 minutes on loss of instrument air. System reliability is demonstrated by five consecutive, cold, quick starts. No damage occurs to the feedwater system from waterhammer.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 57 of 71)

THERMAL EXPANSION TEST SUMMARY

OBJECTIVE

To demonstrate that Reactor Coolant System components and Class 1, 2 and 3 systems and piping systems which experience temperatures of greater than 200°F expand and move without obstruction during heatup and return to ambient. Additionally, to verify that loads and clearance gaps of selected piping system snubbers, variable spring hangers and pipe rupture restraints are properly set and allow the piping to move without obstruction or binding.

PREREQUISITES

1. The piping system which is to be subjected to the thermal expansion has been reviewed to ensure that all hangers which allow or restrict thermal growth have been installed and adjusted.

2. Preliminary shimming has been completed and shim crevices and support surfaces have been inspected to ensure debris has been removed.

3. The required heat load is available to subject the system to its designed thermal expansion.

4. Preservice examination of all snubbers verified completed within 6 months.

TEST METHOD

1. Verify that the selected system can expand without obstruction or interference during system heatup from ambient condition to operating condition.

2. Verify that the Reactor Coolant System piping and components are capable of returning to their cold position within specified tolerance limits.

3. Inspect snubbers and spring cans at specified temperature intervals (250°F, 350°F and 450°F) to ensure their expected thermal movements and swing clearances are such that from cold to hot condition and return to ambient condition are within the criteria per applicable design drawings.

   NOTE: For those systems which do not attain operating temperature, verify that the snubbers and spring cans are set at their calculated cold load position and observe that the system supports are free to move without binding or obstruction.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

ACCEPTANCE CRITERIA

The systems and components defined by FSAR Section 3.9 which have been verified by the thermal expansion tests are free to move without obstruction or binding due to thermal loads. The system supports (i.e., snubbers, spring cans) move within the allowable range per applicable design specifications.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

PRESSURIZER SAFETY AND RELIEF VALVES TEST SUMMARY

OBJECTIVE

To demonstrate the proper operation of the pressurizer power-operated relief valves, the proper functioning of the pressurizer relief tank, and the leaktightness of the pressurizer safety valves.

PREREQUISITES

1. Plant conditions are established as necessary for test performance.

2. The pressurizer relief tank is filled to its normal level and a water supply is available for the relief tank sprays.

TEST METHOD

1. Verify proper actuation and operation of the power-operated relief valves by raising pressurizer pressure above the setpoints with the relief valve isolate.

2. With the Reactor Coolant System at normal operating pressure verify that safety valve leakage is within acceptable limits.

3. Check operation of the discharge header leak detection devices.

4. Verify the ability of the pressurizer relief tank to condense a steam discharge from the pressurizer.

5. Verify proper operation of the pressurizer relief tank level control system, instrumentation, interlocks, and alarms.

6. Confirm that cooling spray to the pressurizer relief tank is initiated as required.

7. The pressurizer safety valves will be hydrostatically bench tested to verify their set points.

ACCEPTANCE CRITERIA

The pressurizer safety valves and power-operated relief valves and the pressurizer relief tank function in accordance with design requirements. Safety valve leakage is within acceptable limits. Controls, instrumentation, interlocks and alarms operate properly in response to simulated or normal input signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

FUEL HANDLING AND VESSEL SERVICING EQUIPMENT TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the fuel handling and vessel servicing equipment, including the handling equipment components and fuel transfer system.

PREREQUISITES

1. The refueling cavity, refueling canal, and spent fuel pool are clean and areas adjacent to the system equipment are clear.

2. Dummy assembly and test weights are available as required for testing the polar crane, manipulator crane, spent fuel pit bridge crane and fuel building overhead crane.

TEST METHOD

1. With the use of a dummy assembly demonstrate the proper operation of all system components, including the manipulator crane, spent fuel pit bridge and electric hoist, new fuel elevator, fuel transfer system, rod cluster control changing fixtures, various handling tools, and indexing of system.

2. Verify operation of interlocks and proper setting of limit switches.

3. Verify leaktightness of sectionalizing devices as necessary.

4. Perform a 125% static load test and a 100% load full operational test on the polar crane, fuel building overhead crane, manipulator crane and the spent fuel pit bridge crane.

ACCEPTANCE CRITERIA

The fuel handling and vessel servicing system components operate in accordance with design requirements. Limit switches are properly set and interlocks function properly, including, as applicable, conforming to technical specifications requirements. Sectionalizing devices meet design requirements for leaktightness.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 61 of 71)

REACTOR COOLANT SYSTEM COLD HYDROSTATIC TEST SUMMARY

OBJECTIVE

To verify the integrity and leak-tightness of the Reactor Coolant System by performing a hydrostatic test of the system, in accordance with Section III of ASME Boiler and Pressure Vessel Code.

PREREQUISITES

1. The reciprocating charging pump is operational.

2. A water supply within acceptable chemistry and temperature limits is available for pressurizing the Reactor Coolant System.

TEST METHOD

1. Pressurize the Reactor Coolant System within the maximum rate and in the prescribed increments until the desired test pressure is obtained, and stabilize at the test pressure for the required time.

2. Perform inspections of the Reactor Coolant System welds, joints, piping, and components.

3. Reduce system pressure to below the test pressure value.

ACCEPTANCE CRITERIA

The cold hydrostatic test satisfactorily verifies the integrity and leak-tightness of the Reactor Coolant System welds.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 62 of 71)

INTEGRATED HOT FUNCTIONAL TEST SUMMARY

OBJECTIVE

To demonstrate the satisfactory performance of systems and components during Reactor Coolant System heatup, at normal operating pressure and temperature, and during Reactor Coolant System cooldown.

PREREQUISITES

1. The RCS Cold Hydrostatic Test has been completed.

2. All systems, or portions of systems, and components whose adequacy or proper operations are to be verified under hot plant conditions, are operational.

3. Reactor vessel internals have been inspected.

TEST METHOD

1. Heat the RCS to normal operating temperatures and pressure utilizing heat from the reactor coolant pumps and pressurizer heaters.

2. Check the thermal expansion of system components and piping. (See Thermal Expansion Test Summary (Sheet 59 of 31))

3. Perform isothermal calibration of Resistance Temperature Detector and incore thermocouples.

4. Verify capability of the Chemical and Volume Control System to provide charging water at rated flow against normal reactor coolant pressure, check letdown design flow rate for each applicable operating mode, and check response of the system changes in pressurizer level.

5. Demonstrate proper operation of the pressurizer relief valves, and verify proper operation of the Pressurizer Relief Tank.

6. Verify proper operation of steam generator instrumentation to changes in steam generator level.

7. Demonstrate proper functioning of the Main Steam Isolation Valves under normal operating pressure and temperature conditions.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

8. Operate the RC pumps for a minimum of 240 hours at full flow in order to achieve greater than one million cycles on vessel internals. Following hot functional testing, the internals are removed and inspected for vibration effects.

9. Perform periodic vibration measurements on RCS components as required.

10. Verify acceptability of the excess letdown and seal water flows.

11. Perform a controlled plant cooldown by using steam dump from the steam generators and operating the Residual Heat Removal System.

12. Demonstrate that the effectiveness of the pressurizer heaters is within acceptable limits.

ACCEPTANCE CRITERIA

The systems and components checked during Hot Functional Testing function in accordance with design specifications and applicable FSAR requirements. Applicable Technical Specification requirements are satisfied.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

OPERATIONAL VIBRATION TESTING TEST SUMMARY

OBJECTIVE

To verify that the vibration level of selected (1) Class 1, 2, and 3 piping (2) other high-energy piping systems inside Seismic Category I Structures, (3) high-energy portions of systems whose failure could reduce the functioning of any Seismic Category I Plant Feature to an unacceptable level is within acceptable levels, for systems requiring test. See FSAR sections 3.9B.2.1.2 and 3.9B.2.1.3.

PREREQUISITES

1. Systems are operational as required.

2. Instrumentation is in place for testing as required.

TEST METHOD

1. Subject the specified piping systems to various flow modes and transients such as pump trips and valve closures as required.

2. Visually inspect and/or measure the vibration level of the piping and components at the specified locations.

3. Following completion of the system transient test, visually inspect the piping and supports including snubbers for damage, looseness of parts etc.

ACCEPTANCE CRITERIA

The vibration level for piping and components are within acceptable limits. For acceptance criteria basis see FSAR Sections 3.9B.2.1.2 and 3.9B.2.1.3.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 65 of 71)

CONTAINMENT LOCAL LEAK RATE TEST SUMMARY

OBJECTIVE

To detect local leaks and measure leakage across each pressure-containing or leakage-limiting boundary for the primary reactor containment penetrations as applicable, including containment isolation valves.

PREREQUISITES

Equipment is available to provide measurement of local leak rate at penetrations, and containment isolation valves.

TEST METHOD

1. Examine the individual containment penetrations and containment isolation valves for leakage per Appendix J of 10 CFR Part 50.

2. Measure the rate of pressure loss across the containment penetration pressure-containing or leakage-limiting boundaries, per Appendix J of 10 CFR Part 50.

ACCEPTANCE CRITERIA

The Containment Local Leak Rate Tests meet the requirement of 10 CFR Part 50, Appendix J.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 66 of 71)

CONTAINMENT INTEGRATED LEAK RATE TEST SUMMARY

OBJECTIVE

To verify the primary reactor containment overall integrated leakage rate is within acceptable limits.

PREREQUISITES

1. Fluid system conditions are established as applicable to simulate post accident conditions which extend the boundary of the Containment Building.

2. Containment component and isolation valve leak tests have been satisfactorily performed.

3. All containment isolation valves have been closed by normal actuation methods.

TEST METHOD

1. Perform the containment integrated leak rate test per Appendix J of 10 CFR Part 50.

2. Perform the leakage rate calculation by using the mass-point methodology as described by ANSI/ANI 56.8-1987 or the methods of ANSI N45.4-1972.

3. If during the performance of a type A test, excessive leakage occurs through locally testable penetrations or isolation valves, these leakage paths may be isolated and the Type A test continued until completion. The sum of the post repaired minimum pathway local leakage rate values will be added to the UCL per ANSI 56.8- 1981.

ACCEPTANCE CRITERIA

The Containment Integrated Leak Rate Test meets the requirements of Appendix J of 10 CFR Part 50.

Note: The containment structural integrity test described in FSAR Section 3.8 may be performed concurrently with the Integrated Leak Rate Test.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 67 of 71)

CONTAINMENT ISOLATION SYSTEM TEST SUMMARY

OBJECTIVE

To verify that the containment isolation valves close as required on a containment isolation signal.

PREREQUISITES

1. Plant conditions are established as necessary for test performance.

TEST METHOD

1. Verify that the containment isolation valves close properly upon receipt of a containment isolation signal.

ACCEPTANCE CRITERIA

The Containment Isolation System functions in accordance with FSAR specifications.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 68 of 71)

REACTOR VESSEL WATER LEVEL INDICATION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Reactor Vessel Water Level Indication system to detect the approach to inadequate core cooling by indication of collapsed water level in the upper head and upper plenum regions of the Reactor Vessel.

PREREQUISITES

1. Reactor Vessel Head is installed and Heated Junction Thermocouple Probes are in place.

2. A temporary external means of determining actual Reactor Vessel water level is installed.

3. The ERF Computer is operational.

4. Plant conditions are established as required to allow control of Reactor Vessel water level.

TEST METHOD

1. Verify that Reactor Vessel water level changes inside the range of the Heated Junction Thermocouple Probes are indicated sequentially on the control board and the ERF computer.

2. Verify that a correlation exists between actual Reactor Vessel water level and indicated level.

ACCEPTANCE CRITERIA

The Reactor Vessel Water Level Indication System responds properly to water levels inside its measurement range. Redundant instrumentation channels indicate agreement with actual water level.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 69 of 71)

ANTICIPATED TRANSIENT WITHOUT SCRAM MITIGATION SYSTEM ACTUATION CIRCUITRY TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the anticipated transient without scram mitigation system actuation circuitry to respond properly to logic initiation signals.

PREREQUISITE

1. Prior to fuel load.

TEST METHOD

1. Demonstrate proper operation of the anticipated transient without scram mitigation system actuation circuitry under various logic conditions.

2. Perform tests utilizing signals or simulated signals on each of the anticipated transient without scram mitigation system actuation circuitry inputs in accordance with the applicable manufacturer's instruction manual.

ACCEPTANCE CRITERIA

The anticipated transient without scram mitigation system actuation circuitry functions in accordance with design specifications in response to logic initiation signals.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 70 of 71)

SAFETY RELATED AIR ACCUMULATOR TEST SUMMARY

OBJECTIVE

To verify the ability of safety related components with backup air accumulators to perform their function on a rapid or gradual loss of the instrument air supply.

PREREQUISITES

The Instrument Air System is operational and aligned in accordance with test documents.

TEST METHOD

1. Subject the air accumulators to a rapid loss of air supply upstream of the check valves to demonstrate the ability of the check valves to seat properly.

2. Subject the air accumulators to a gradual loss of air supply upstream of the check valves to demonstrate the ability of the check valves to seat properly.

3. Perform a test of the air operated components supplied by accumulators to demonstrate their ability to perform their design basis function.

ACCEPTANCE CRITERIA

The check valves seat on both a gradual and rapid loss of instrument air supply. The accumulators have sufficient capacity, including steady state consumption and possible leakage, to supply air to the device so that it can perform its design basis function.

TABLE 14.2-2
PREOPERATIONAL TESTS SUMMARIES INDEX

(Sheet 71 of 71)

INSTRUMENT AIR SYSTEM BRANCH HEADER TEST SUMMARY

OBJECTIVE

To verify that all air operated components supplied by branches of the Instrument Air System which supply safety related accumulators will respond as expected to a loss of air supply.

PREREQUISITES

1. The Instrument Air System is operational and aligned in accordance with test documents.

2. Air operated components of the branch to be tested are in the proper position in accordance with test documents.

TEST METHOD

Isolate the branch header from the air supply and allow the branch header pressure to decay through normal steady state consumption of air operated components.

ACCEPTANCE CRITERIA

All air operated components of the branch respond as expected. Accumulator check valves seat properly.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

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TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 2 of 28)

REACTOR COOLANT SYSTEM FLOW TEST SUMMARY

OBJECTIVE

To verify predicted Reactor Coolant System cold leg volumetric flow rates at normal operating temperature and pressure with all reactor coolant pumps running in hot standby and during power ascension testing and demonstrate that pressurizer spray is within acceptable limits.

PREREQUISITES

1. The reactor is at the specified power level.

2. The RCS is at the specified conditions.

3. All reactor coolant pumps are operational.

TEST METHOD

1. During hot standby operation, measure and record loop elbow differential pressures and determine cold leg volumetric flow rates. At 50% and 75% power for Unit 1 and 75% and 100% power for Unit 2 use the N-16 Transit Time Flow Meter and a precision secondary calorimetric to determine loop cold leg volumetric flow rates.

2. Verify that the reactor coolant system flow transmitters have been aligned for zero flow and 100 percent flow at normal operating conditions.

3. Demonstrate that the effectiveness of the pressurizer spray is within acceptable limits.

ACCEPTANCE CRITERIA

The measured Reactor Coolant System flow is within design flow limits specified in FSAR Chapter 5, and the flow transmitters are satisfactorily aligned for zero flow and full flow conditions. Pressurizer spray is within acceptable limits.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 3 of 28)

REACTOR COOLANT SYSTEM FLOW COASTDOWN TEST SUMMARY

OBJECTIVE

To measure the reactor coolant system flow rate decrease subsequent to a simultaneous trip of all four reactor coolant pumps, and to measure the delay times associated with assumptions of the loss of flow accident analysis.

PREREQUISITES

1. The reactor is in the hot standby condition.

2. Applicable portions of the Reactor Coolant System Flow Test have been satisfactorily completed, and any pressure damping devices, if installed for that test, have been removed.

TEST METHOD

1. With the reactor coolant system in the hot standby condition, simultaneously trip all four reactor coolant pumps and measure the rate at which reactor coolant flow decreases.

2. Determine the delay times associated with the low flow reactor trip circuitry.

ACCEPTANCE CRITERIA

The rate of reactor coolant flow decrease, upon tripping of all four reactor coolant pumps, satisfies the NSSS vendor acceptance criteria which verifies the loss of flow analysis of the FSAR, and the associated delay times are within acceptable limits.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 4 of 28)

CONTROL ROD DRIVE TEST SUMMARY

OBJECTIVE

To verify control rod bank start and stop setpoints, verify proper slave cycler timing and drive mechanism operation, check rod speeds, and demonstrate the capability of the CRDMs to respond to signals from the Reactor Control System.

PREREQUISITES

1. The rod cluster control assemblies, control rod drive mechanisms and Reactor Control System are operational.

2. Plant conditions are established as necessary.

TEST METHOD

1. Verify proper rod bank start and stop positions during rod insertion and withdrawal, and check the Rod Control System bank- overlap setpoints and rod speeds.

2. Demonstrate proper slave cycler timing and drive mechanism operation.

3. Demonstrate proper functioning of the Rod Control System during steady state critical operations, and observe the automatic response of the CRDM’s to signals from the Reactor Control System.

ACCEPTANCE CRITERIA

All mechanical operating features of the rod drive mechanisms and associated rod cluster control assemblies function properly, rod speeds are within acceptable limits, and the CRDM’s satisfactorily respond to signals from the Reactor Control System.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 5 of 28)

ROD POSITION INDICATION TEST SUMMARY

OBJECTIVE

To verify that the Digital Rod Position Indication system provides proper rod position indication and alarms based on simulated and/or actual inputs over the entire length of travel of each rod cluster control assembly.

PREREQUISITES

1. The full-length rod cluster control assemblies are prepared for operation.

2. The Digital Rod Position Indication system is energized and properly aligned.

TEST METHOD

1. Verify proper response of the Digital Rod Position Indication system over the entire length of travel of the rod cluster control assemblies.

2. Verify proper functioning of the indication and alarm features associated with the rod position indication circuitry.

ACCEPTANCE CRITERIA

The rod position indicators perform satisfactorily for each full- length rod cluster control assembly over its entire length of travel. The rod-bottom alarms function in accordance with design specifications. Indication accuracy and alarm setpoints are within design criteria.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 6 of 28)

REACTOR TRIP SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the proper functioning of the Reactor Trip System, including the capability to test the operation of the reactor trip breakers using bypass breakers.

PREREQUISITES

1. Fuel loading is complete.

2. Electrical power is available for the Reactor Trip System circuitry.

3. Control rod drive mechanisms are operational (applies only to the test method #1 unlatching test).

TEST METHOD

1. Demonstrate that each control rod drive mechanism will properly unlatch upon opening of the trip breakers.

2. Confirm that providing a simulated trip signal to test a reactor trip breaker, with its associated bypass breaker closed, will open the reactor trip breaker without opening its associated bypass breaker.

3. Demonstrate the capability to manually open the trip breakers.

4. With all control rods fully inserted attempt to close both reactor trip breaker bypass breakers and verify interlocks cause both bypass breakers to trip. With one reactor trip breaker bypass breaker closed, verify that by placing the opposite trip channel in test causes both reactor trip breakers and both bypass breakers to open.

ACCEPTANCE CRITERIA

Each control rod drive mechanism unlatches upon opening of the trip breakers. The associated reactor trip breaker bypass breaker remains closed when each reactor trip breaker is opened for test. Interlocks which prevent closing both reactor trip breaker bypass breakers simultaneously function in accordance with design requirements.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 7 of 28)

AUXILIARY STARTUP INSTRUMENTATION TEST SUMMARY

OBJECTIVE

To demonstrate the proper response of the temporary neutron detectors and proper functioning of their associated indicating and recording functions.

PREREQUISITE

1. A neutron source is available.

TEST METHOD

1. With the use of the neutron source, verify that the auxiliary startup instrumentation responds properly with indication and an optional audible signal.

2. Verify that an increase in the indicated count rate occurs during the initial stages of fuel load.

ACCEPTANCE CRITERIA

The auxiliary startup instrumentation responds properly to a neutron source providing indication and an optional audible signal. The instrumentation indicates increasing neutron levels during early stages of fuel loading.

Note: Refer also to Section 14.2.10.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 8 of 28)

CALIBRATION OF PROCESS TEMPERATURE AND NUCLEAR INSTRUMENTATION
TEST SUMMARY

OBJECTIVE

To calibrate and adjust the operational settings of the source, intermediate, and power range detectors. To verify channel indication overlap and power range detector output linearity. Also, to calibrate and adjust the operational settings of the N-16 power detectors and reactor coolant average temperature instrumentation system.

PREREQUISITES

1. The preoperational tests of the nuclear and temperature instrumentation systems have been satisfactorily completed.

2. Reactor power level is established as necessary.

TEST METHOD

1. Determine the source range channel voltages versus detector output and readjust as necessary prior to initial fuel load.

2. Perform an isothermal alignment of the N-16 power and T_{ave} instrumentation prior to initial criticality.

3. Measure overlap between the source range, intermediate range, and power range channels.

4. At each major power plateau, determine reactor power by secondary calorimetric and calibrate each power range excore NIS and each N-16 power channel to correspond to the measured thermal power. Also, adjust each N-16 T_{ave} channel as necessary.

5. Verify acceptability of each power range channel output by measuring and plotting power range detector currents versus power level. Determine the linearity or each power range channel and the degree of uniformity between channels.

6. Verify proper functioning of the nuclear instrumentation indication and selected alarm features.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 9 of 28)

ACCEPTANCE CRITERIA

The source range channel operating voltages are set for optimum neutron response. The source, intermediate, and power range channels exhibit satisfactory overlap. The power range excore detectors display a linear output over the range of normal power operation, and accurately indicate the actual reactor power level as determined by calorimetric measurements. Selected alarm features function properly. The N-16 power and T_{ave} channels accurately indicate the actual reactor power and reactor coolant average temperature as determined by calorimetric measurements.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 10 of 28)

CHEMICAL TEST SUMMARY

OBJECTIVE

To establish the water chemistry of the Reactor Coolant System, and to verify the capability to maintain Reactor Coolant System chemistry at power and during power escalation.

PREREQUISITES

1. The chemical addition portion of the CVCS is operational.

2. Sampling analysis equipment is operational.

TEST METHOD

1. Prior to fuel load, establish the Reactor Coolant System water chemistry within specifications.

2. Verify that the water chemistry can be maintained within specifications during criticality and power escalation.

ACCEPTANCE CRITERIA

Reactor Coolant System water chemistry is satisfactorily established prior to fuel load, and maintained within specifications during criticality and power escalation.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 11 of 28)

RADIATION SURVEYS TEST SUMMARY

OBJECTIVE

To verify radiation shielding effectiveness by measuring radiation dose levels at preselected locations within the plant during low, intermediate and high reactor power level operation.

PREREQUISITES

1. The radiation survey instruments are calibrated against known sources.

2. Reactor power level is established as necessary.

TEST METHOD

1. At specified steady state power levels between zero and 100 percent power, measure radiation levels at preselected locations within the plant to determine effectiveness of the radiation shielding.

2. Upon completion of the radiation surveys, check the calibration of the survey instruments.

ACCEPTANCE CRITERIA

The measured radiation levels are within limits for the zone designation of each area surveyed.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 12 of 28)

PROCESS AND EFFLUENT RADIATION MONITORING TEST SUMMARY
(UNIT 1)

OBJECTIVE

To verify the proper performance of the process and effluent radiation monitoring equipment under actual nuclear operating conditions.

PREREQUISITES

1. The process and effluent radiation monitors have been checked against known sources.

2. The reactor has been operating at power for a time sufficient to provide representative process and effluents samples.

TEST METHOD

1. By radiochemical analysis verify the suitability of liquid effluents for discharge.

2. When conditions permit, perform an actual liquid discharge and observe the response of the effluent monitors.

3. By radiochemical analysis of the process and effluent samples confirm the satisfactory performance of the process and effluent monitors.

ACCEPTANCE CRITERIA

The installed process and effluent monitors properly indicate the radioactive content of the fluid monitored and perform in accordance with design specifications.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 13 of 28)

PROCESS AND EFFLUENT RADIATION MONITORING TEST SUMMARY
(UNIT 2)

This test conducted during preoperational testing

(See Table 14.2-2)

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

MODERATOR TEMPERATURE REACTIVITY COEFFICIENT TEST SUMMARY

OBJECTIVE

To confirm that the actual moderator temperature coefficient of reactivity is within acceptable limits.

PREREQUISITES

1. The reactor is critical, in the hot, zero power condition.

TEST METHOD

1. Initiate a small change in temperature of the Reactor Coolant System.

2. Determine the amount of reactivity inserted or removed by the temperature change, and calculate the value of the moderator temperature coefficient.

ACCEPTANCE CRITERIA

The moderator temperature coefficient of reactivity is determined to be within acceptable limits.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 15 of 28)

CONTROL ROD REACTIVITY WORTHS TEST SUMMARY

OBJECTIVE

To verify the design rod worths of the control and shutdown RCCA banks.

PREREQUISITES

1. The reactor is critical, in the hot, zero power condition.

TEST METHOD

1. With the use of boron concentration sampling data, rod position indication and reactivity measurement, verify the worths of the control and shutdown RCCA banks utilizing the method of Bank Exchange (or Boron Concentration Exchange).

ACCEPTANCE CRITERIA

The rod worths are determined to be within design specifications.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

BORON REACTIVITY WORTH TEST SUMMARY

OBJECTIVE

To measure the reactivity worth of the boron in the reactor coolant.

PREREQUISITES

1. The reactor is critical, in a hot, zero power condition.

TEST METHOD

1. Determine the reactivity worth of the boron in solution with the reactor coolant by changing the boron concentration, and compensating for the reactivity effect by inserting or removing reactivity with control rod motion.

2. Equate the change in reactivity due to rod motion to the worth of the change in boron concentration.

ACCEPTANCE CRITERIA

The value of the boron reactivity worth is determined to be within acceptable limits.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

CORE REACTIVITY BALANCE TEST SUMMARY

OBJECTIVE

To verify that actual core reactivity effects are in agreement with design values.

PREREQUISITES

1. The reactor power is established as necessary.

TEST METHOD

1. Measure the all rods out, hot zero power, xenon free, critical boron concentration.
2. Measure the all rods out, hot full power, equilibrium xenon, critical boron concentration.
3. Perform a core reactivity balance.

ACCEPTANCE CRITERIA

The overall core reactivity balance is within acceptable limits.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

LOSS OF OFFSITE POWER TEST SUMMARY

OBJECTIVE

To demonstrate the proper plant response following a plant trip with no offsite power available.

PREREQUISITES

The Turbine-Generator output is approximately 130 MWe (greater than 10% reactor power) with non-Class 1E buses being supplied from the unit auxiliary transformer and the Class 1E buses being supplied from their offsite power source.

TEST METHOD

1. Manually generate a main turbine trip and isolate offsite power sources.

2. Verify proper starting and load sequencing of diesel generators and transfer of power supplies for all required equipment.

3. Verify the Reactor Coolant System can be maintained in a shutdown condition for a minimum of 30 minutes utilizing the power operated atmospheric relief valves to remove decay heat from the reactor core.

ACCEPTANCE CRITERIA

The on-site power supplies (i.e., diesel generators) shall auto-start and operate the necessary controls, equipment and indication to remove decay heat and maintain the Reactor Coolant System in a shutdown condition for the duration of the test.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 19 of 28)

ROD DROP TEST SUMMARY

OBJECTIVE

To determine the rod drop time of each full-length rod cluster control assembly (RCCA) under hot full flow Reactor Coolant System conditions.

PREREQUISITES

1. The Reactor Coolant System is at nominal, no load, operating temperature and pressure, prior to initial criticality.

2. Rod position indication is functional.

3. All four Reactor Coolant Pumps are operating.

TEST METHOD

1. Withdraw each full-length rod cluster control assembly, interrupt the electrical power to the associated rod drive mechanism, and measure and record the rod drop time. This test is performed with the reactor in hot full flow conditions.

2. Perform at least three additional rod drop tests for each rod whose measured drop time deviates from the mean for all rods by more than two standard deviations.

   NOTE: Additional rod drop tests may be performed at the option of the test engineer. These optional tests are not used to satisfy acceptance criteria and may be performed under hot or cold RCS conditions with any RCS flow condition.

ACCEPTANCE CRITERIA

The hot, full flow rod drop times are acceptable in accordance with plant Technical Specifications.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

FLUX DISTRIBUTION MEASUREMENTS TEST SUMMARY

OBJECTIVE

To determine the reactor core power distribution.

PREREQUISITES

1. Incore instrumentation and process computer are operable for incore flux mapping.

2. Reactor is critical and power level is established as necessary.

TEST METHOD

Complete an incore flux map for the All Rods Out (ARO) control rod configurations with reactor power stabilized below 5 percent, for Unit 1 and prior to exceeding 30 percent for Unit 2.

Note: ARO is defined for this measurement as Control Bank D above 190 steps withdrawn and all other banks fully withdrawn.

ACCEPTANCE CRITERIA

The core flux distributions indicated by the flux map are acceptable in accordance with plant Technical Specifications where applicable.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 21 of 28)

CORE PERFORMANCE EVALUATION TEST SUMMARY

OBJECTIVE

To verify the operating characteristics of the core and the calibration of the flux and temperature instrumentation during power escalation.

PREREQUISITES

1. Reactor power level is established as necessary.

TEST METHOD

1. At steady state power levels of 30, 50, 75, 90 and 100 percent, record Reactor Coolant System parameters. Incore data for flux maps are recorded at 50, 75 and 100 percent.

2. Analyze the data obtained at each power level to determine core performance margins and verify flux and temperature instrumentation calibration.

ACCEPTANCE CRITERIA

The core performance margins are within design predictions for normal rod configurations and the calibration of the flux and temperature instrumentation has been verified.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 22 of 28)

UNIT LOAD TRANSIENTS TEST SUMMARY

OBJECTIVE

To demonstrate satisfactory plant transient response to various specified load changes and trips, to monitor the behavior of reactor control systems during these transients, and, if necessary, optimize the reactor control system setpoints.

PREREQUISITES

1. Reactor power level is established as necessary for each transient.

2. All reactor control systems are operational and their setpoints have been set to their recommended values.

TEST METHOD

1. Initiate a step change in power level of 10 percent and monitor Reactor Coolant System behavior in response to the transients. For Unit 1, this test will be performed at approximate power levels of 50 percent, 30 percent (following completion of 50 percent testing) and 100 percent. For Unit 2, this test will be performed at approximate power levels of 50 percent and 75 percent.

2. Monitor plant response to a 50 percent load reduction, from power levels of approximately 75 percent (Unit 1 only) and 100 percent.

3. Monitor plant response to a plant trip from power levels up to 100 percent.

4. If necessary, adjust the reactor control system setpoints until optimal response is obtained during subsequent test performance.

ACCEPTANCE CRITERIA

Plant response to the unit load transients is acceptable in accordance with design specifications, and the Reactor Control System parameters reach steady state values without appreciable overshoot or oscillation subsequent to a step change.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 23 of 28)

REMOTE SHUTDOWN TEST SUMMARY

OBJECTIVE

To demonstrate the capability of performing a safe plant shutdown, maintain the plant in a hot standby condition, and to demonstrate the ability to cooldown from hot standby to cold shutdown conditions from outside the control room, using the minimum shift crew. Verify that the Remote Shutdown Panel selector switches properly transfer control from the Control Room to the Remote Shutdown panel.

PREREQUISITES

1. The equipment and instrumentation associated with the Remote Shutdown Panel are available for achieving and maintaining the plant in a hot standby condition.

2. The plant is at a power level greater than 10% generator power but less than 25% reactor power, for the reactor trip portion of the test.

3. For the cooldown portion, the plant is in a stable hot standby condition.

TEST METHOD

1. With the generator at greater than 10 percent power, perform a safe shutdown of the plant from outside the Control Room using the minimum shift crew.

2. Check functioning of instrumentation, controls, interlocks and alarms. Credit may be taken for preop/prereq functional tests.

3. Demonstrate the capability to achieve and maintain the plant in a hot standby condition from the Remote shutdown panel for a minimum of 30 minutes.

4. Demonstrate the potential for cooldown to cold shutdown conditions by placing the residual heat removal system into service and reducing the reactor coolant temperature to approximately 300°F.

ACCEPTANCE CRITERIA

Transfer of control to outside the Control Room can be achieved in accordance with design requirements, remote shutdown instrumentation, controls, alarms and interlocks function properly. The potential ability to perform a safe shutdown, to achieve and maintain hot standby conditions from outside the Control Room has been demonstrated. The potential ability to cool down to cold shutdown conditions from outside the control room has been demonstrated.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 24 of 28)

TURBINE TRIP/GENERATOR LOAD REJECTION TEST SUMMARY

OBJECTIVE

To demonstrate the ability of the plant to sustain a full load rejection of the turbine generator at full power, and to evaluate plant response to the transient.

PREREQUISITES

1. Reactor is at a steady state, approximately 100% power condition.

2. Electrical distribution system is aligned for normal power operation.

3. Recording devices are available to record the following parameters; RCS pressure and temperature, pressurizer level, steam pressure, power level and turbine speed.

TEST METHOD

1. With the plant at approximately 100 percent power, manually open the generator main breakers and, observe the effects of the resultant plant trip, and record values of pertinent parameters during the transient.

2. Observe the response of the plant control systems during the transient.

ACCEPTANCE CRITERIA

A Loss of Load trip from full power conditions and resultant plant trip is sustained and the control systems function properly within specified limits to preclude lifting of the pressurizer or main steam safety valves, and subsequent to the trip, the plant can be maintained in a hot standby condition.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 25 of 28)

REACTOR COOLANT LEAK TEST SUMMARY

OBJECTIVE

To demonstrate the leak tightness of Reactor Coolant System pressure boundary and of the reactor vessel flange after the system has been closed following fueling.

To determine the leak rate for primary-to-secondary leakage, Reactor Coolant Pump seal leakage, other identified leakage, and any unidentified leakage.

PREREQUISITES

1. At least one RC pump is operating.

2. Normal operating pressure is being maintained.

3. RC temperature is between 400°F and normal no-load average temperature.

4. The reactor is shutdown prior to initial criticality.

TEST METHOD

1. Verify there is no vessel flange leakoff.

2. Verify Reactor Coolant System integrity by visual inspection.

3. Determine the primary-to-secondary leak rate by sampling the secondary for boron.

4. Determine the Reactor Coolant Pump seal leak rate.

5. Obtain results of leakage rates testing of Reactor Coolant Pressure Isolation Valves.

6. Determine the identified and unidentified leakage rate by conducting a mass balance of the primary system.

ACCEPTANCE CRITERIA

1. There is no indication of vessel flange leakage.

2. Visual inspection of the Reactor Coolant System is satisfactory.

3. The leakage rates are within Technical Specification limitations.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 26 of 28)

ROD CONTROL SYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate that the Full Length Rod Control System performs the required control and indication functions in order to verify it is operational and ready for use prior to initial criticality.

PREREQUISITES

1. All rod drop tests and mechanism timing alignment is complete and verified to be within specifications.

2. Nuclear Instrumentation Source Range channels shall be aligned and operable and shall be monitored at all times during conduct of this test.

3. The plant is at normal no load operating temperature and pressure prior to initial criticality.

4. The digital Rod Position Indication System is operable and in service.

5. The boron concentration is being maintained at refueling shutdown concentration.

6. The bank overlap settings are set to allow overlap operation of all control banks and still maintain a minimum amount of rod withdrawal.

TEST METHOD

1. The manual mode of control is checked for each applicable position of the bank selector switch.

2. Rod speed and direction are verified to be in accordance with manufacturer’s instruction manual.

3. Verify proper operation of status lights, step counters, rod position and speed indications.

4. Verify proper operation of rod bank overlap.

ACCEPTANCE CRITERIA

The Rod Control System responds properly to normal input signals in the manual mode, providing correct bank overlapping, rod speed, direction and indication.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 27 of 28)

AUTOMATIC REACTOR CONTROL SYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate the ability of the Automatic Reactor Control System to return reactor coolant temperature to the programmed setpoint and to maintain it.

PREREQUISITES

1. The reactor is at approximately 50% power.

2. Reactor Coolant System temperature and pressure are stable.

3. Pressurizer pressure and level control are in automatic.

4. Steam Generator level control is in automatic.

TEST METHOD

1. At 50% power, raise Reactor Coolant System temperature and observe the Automatic Reactor Control System response. Then lower Reactor Coolant System temperature and observe the Automatic Reactor Control System response.

ACCEPTANCE CRITERIA

The Reactor Coolant System Temperature (T_{avg}) returns to within 1.5°F of the setpoint (T_{ref}) following the positive and negative temperature transient.

TABLE 14.2-3
INITIAL STARTUP TEST SUMMARIES

(Sheet 28 of 28)

INCORE NUCLEAR INSTRUMENTATION TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Incore Nuclear Instrumentation to remotely position the incore neutron detectors for the purpose of core flux mapping, and to supply the appropriate digital and analog signals to the plant computer.

PREREQUISITES

1. Fuel has been loaded in the core and all thimbles have been inserted and sealed.

2. Dummy cables and other equipment necessary for the test performance are available.

TEST METHOD

1. Ensure free passage and index the system by driving a dummy cable to all thimble positions and determine preliminary limit switch settings.

2. Verify operation of 5-path and 10-path transfer assemblies in all modes using a dummy cable.

3. Provide a check of the leak detection system.

4. During flux mapping at power, verify detector response to neutron flux and make final limit switch settings.

5. Verify each detector/cable combination can be inserted to the top of core limit switches, automatically stops and can be withdrawn.

ACCEPTANCE CRITERIA

The Incore Nuclear Instrumentation channels function properly and supply appropriate outputs to the plant computer. The indexing system functions in accordance with design requirements. After fuel loading, free passage to all positions is demonstrated. Response to neutron flux at power meets design criteria.