Enclosure 1
Changes to PSAR Chapter 3 and 9
(Non-Proprietary)

Table 3.1-2: Principal Design Criteria

Principal Design Criteria	SAR Section
PDC 1, Quality Standards and Records	3.5, 4.3, 6.3, 7.3, 7.4, 7.5
PDC 2, Design bases for protection against natural phenomena	3.5, 4.2.2, 4.3, 4.7, 5.1, 6.3, 7.3, 7.4, 7.5, 8.2, 8.3, 9.1.1, 9.1.2, 9.1.3, 9.1.4, 9.1.5, 9.2, 9.3, 9.4, 9.7, 9.8.2, 9.8.4, 9.8.5, 11.2
PDC 3, Fire Protection	6.3, 7.3, 7.5, 9.3, 9.4
PDC 4, Environmental and dynamic effects design bases	3.5, 4.2.2, 4.3, 4.7, 6.3, 7.3, 9.1.1, 9.1.2, 9.1.4, 9.3, 9.7, 9.8.2, 9.8.4, 9.9
PDC 5, Sharing of structures, systems, and components	3.1.1
PDC 10, Reactor Design	4.2.1, 4.3, 4.5, 4.6, 5.1, 6.3, 7.3
PDC 11, Reactor Inherent Protection	4.5
PDC 12, Suppression of reactor power oscillations	4.5, 4.6, 5.1
PDC 13, Instrumentation and Control	7.2, 7.3, 7.5, 9.1.3
PDC 14, Reactor Coolant Boundary	4.3
PDC 15, Reactor coolant system design	7.3
PDC 16, Containment design	4.2.1, 5.1
PDC 17, Electric Power systems	8.2, 8.3
PDC 18, Inspection and testing of electric power systems	8.2, 8.3
PDC 19, Control room	7.4
PDC 20, Protection system functions	7.3
PDC 21, Protection system reliability and testability	7.3, 7.5
PDC 22, Protection System Independence	7.3, 7.5
PDC 23, Protection system failure modes	4.2.2, 7.3
PDC 24, Separation of protection and control systems	7.3, 7.5
PDC 25, Protection system requirements for reactivity control malfunctions	7.3

- Protection of safety-related SSCs from design basis natural phenomena and external hazards
- Structural support for safety-related SSCs located on the safety-related portion of the Reactor Building
- Protection from adverse effects of non-safety related SSCs failures on the ability of safety-related SSCs to perform their safety functions
- Prevent interactions between reactor coolant (Flibe) and water contained in concrete in the safety-related portion of the reactor building.

3.5.2 Design Bases

- Consistent with PDC 1, the safety-related portion of the Reactor Building is designed in accordance with industry codes and standards, and the quality assurance program described in Section 12.9.
- Consistent with PDC 2, the safety-related portion of the Reactor Building is designed to provide protection for safety-related SSCs housed within to perform their safety functions in design basis meteorological, water, and seismic events as described in Sections 3.2, 3.3, and 3.4.
- Consistent with PDC 3, the safety-related portion of the Reactor Building is designed with design features to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.
- Consistent with PDC 4, the Reactor Building is designed to be protected from and provide protection of other safety-related SSCs against environmental and dynamic effects associated with high-pressure steam system pipe leaks and breaks.
- Consistent with PDC 75, the Reactor Building is designed to protect the geometry of the decay heat removal system from postulated natural phenomena events.
- Consistent with PDC 76, the Reactor Building is designed to permit appropriate periodic inspection and surveillance of safety-related structural areas.

3.5.3 System Evaluation

Although the non-safety related portion of the Reactor Building surrounds the safety-related portion of the Reactor building, the non-safety related portion is not credited in the safety analysis. Neither the safety-related nor non-safety related portion of the Reactor Building is credited in the safety analysis to perform a safety-related containment function for retention of fission products since the design relies on a functional containment concept (see Chapter 13). Similarly, the non-safety related portion of the Reactor Building is not credited to provide physical protection to safety-related SSCs from the effects of normal or high winds (see Section 3.5.3.1), or from the effects of design basis earthquakes (see Section 3.5.3.3). Finally, the non-safety related portion of the reactor building is not credited to provide protection to safety-related SSCs from the effects of water damage (see Section 3.5.3.2). However, the shape of the exterior roof precludes adverse effects related to accumulation of water and ice. A list of load combinations for the safety-related portion of the Reactor Building is provided in Table 3.5-1.

Consistent with PDC 1, the safety-related portion of the reactor building is under the quality assurance program described in Chapter 12. The safety-related portion of the Reactor Building is designed to the local building code, ASCE/SEI 7-10 (Reference 1), and augmented for specific design basis natural phenomena as described below. The non-safety related portion of the Reactor Building is designed to local building codes which invoke ASCE/SEI 7-10.

Consistent with PDC 3, the safety-related portion of the Reactor Building is designed to perform its safety function in the event of a fire hazard. The safety-related portion of the Reactor Building includes design features which minimize the probability and effect of fires and explosions by the use of low combustible materials and physical separation. These design features, in conjunction with the fire protection program described in Section 9.4, provide assurance that the safety-related portion of the Reactor Building conforms to PDC 3.

Consistent with PDC 4, the safety-related portion of the Reactor Building is designed so it will be able to perform its physical protection safety functions described in Section 3.5.1 under the environmental and dynamic effects associated with high-pressure steam system pipe leaks and breaks, including if the non-safety related portion of the Reactor Building is damaged due to such effects.

The decay heat removal system (DHRS) contains safety-related SSCs, which are located in the safety-related portion of the Reactor Building. The design of the safety-related portion of the Reactor Building protects the safety-related SSCs within it from adverse effects on those safety-related SSCs from design basis natural phenomena described in Sections 3.2, 3.3, and 3.4. This satisfies PDC 75.

The safety-related portion of the Reactor Building is designed to permit appropriate periodic inspection and surveillance. This includes the basement area containing the base isolation system, which is a safety related SSC. This satisfies PDC 76.

Consistent with PDC 2, the safety-related portion of the reactor building is designed so that it will be able to perform its physical protection safety functions described in Section 3.5.1, even if the non-safety related portion of the reactor building is damaged due to the design basis wind, water or earthquake events described in Sections 3.2, 3.3, and 3.4. The system evaluation for PDC 2 is provided in the following subsections.

3.5.3.1 Conformance with PDC 2 for Meteorological Events

Section 3.2.1 describes the normal wind loads used as design parameters for the safety-related portion of the reactor building. Loads from normal winds are in the form of velocity pressure. Section 3.2.2 and Section 3.2.3 describe the high wind loads from tornadoes and hurricanes used as design parameters for the safety-related portion of the reactor building. Loads from high winds are in the form of velocity pressure, atmospheric pressure change, and tornado and hurricane missile impacts. Finally, Section 3.2.4 describes the snow loads used as design parameters for the safety-related portion of the reactor building.

Consistent with PDC 2, the safety-related SSCs are located in the safety-related portion of the Reactor Building which is designed to protect safety-related SSCs from the effects of design basis normal and high winds and snow. The safety-related portion of the Reactor Building is a reinforced concrete structure designed to meet American Concrete Institute (ACI) 349-2013 (Reference 2) with internal safety-related steel structures designed in accordance with ANSI and American Institute of Steel Construction (AISC) Standard N690-18 (Reference 3). Both ACI 349-2013 and AISC N690-18 are standards specific to the design of safety-related nuclear structures and have built-in margin. ACI 349 and ANSI/AISC N690-18 are used to design a structure that can withstand the loads from Section 3.2. By designing the safety-related portion of the Reactor Building in accordance with these two standards, the safety-related portion of the Reactor Building satisfies PDC 2 for design basis loads from normal winds, high winds, and snow, as discussed in Section 3.2.

3.5.3.2 Conformance with PDC 2 for Internal and External Flooding

This section describes how the design basis for the safety-related portion of the Reactor Building, with respect to water damage (internal and external flooding), provides reasonable assurance that potential water damage will not preclude safety-related SSCs from performing their safety-related functions. Section 3.3 characterized the design basis loads related to external and internal flooding postulated events. This section describes how the safety-related portion of the Reactor Building is designed to address those loads.

9.9.1 Steam System

The primary function of the steam system is to utilize the heat from the intermediate heat transport system to generate steam for the turbine generator system, as shown in Figure 9.9-1. The steam system utilizes a regenerative direct-contact evaporator to produce saturated steam by mixing feedwater with a portion of recirculated superheated steam. A small portion of the recirculated superheated steam is injected prior to the steam circulating pump to reduce moisture content to the superheater inlet header. The steam pump directs the saturated steam to the superheaters in each unit's IHTS.

The superheater uses heat from the intermediate heat transport system to superheat the saturated steam, as discussed in Section 5.2. The unit specific superheater outlet steam lines combine the output of the superheaters from each unit into a common header. From the superheater outlet header, the steam system directs about thirty percent of the superheated steam to the turbine generator system and recirculates about seventy percent of the superheated steam. The recirculated superheated steam is used to evaporate feedwater, reduce moisture carry-over from the evaporator, and evaporate excess liquid discharge in the flash vessel.

The turbine generator includes steam extraction lines which are directed to the deaerator and feedwater heater in the feedwater system to raise feedwater temperature prior to entry into the evaporator.

The evaporator is equipped with a blowdown line to remove liquid contaminants. The water blowdown from the evaporator, along with the drains of the feedwater and condensate system, is directed to a flash vessel. The flash vessel uses recirculated superheated steam to evaporate the remaining liquid discharge.

The steam system includes piping, pumps, valves, vessels, drains, and traps and is designed to industry codes and standards, as noted in Table 3.6-2. Unit-specific main steam isolation valves are provided upstream of the superheater outlet header to isolate the steam supply from each unit, as needed, to support single unit operation. The steam system is designed to handle a turbine trip without a corresponding reactor trip via the turbine bypass line and the condenser (which is sized to handle 100% steam load), and the steam relief valves (which have the capability to reject 100% load to the atmosphere). These features are not safety-related.

9.9.1.1 Design Bases

Consistent with PDC 4, nearby safety-related systems are protected against dynamic effects associated with high-pressure steam system pipe leaks and breaks.

Consistent with PDC 60, the design of the steam system supports the control of radioactive materials release during normal reactor operations.

Consistent with PDC 64, the steam system is designed to monitor radioactive releases.

Consistent with 10 CFR 20.1406, the steam system is designed to minimize contamination of the facility and the environment and facilitate eventual decommissioning.

9.9.1.2 System Evaluation

Portions of the steam line piping are located in the non-safety related portion of the reactor building and in the turbine building. Piping is not located near safety-related SSCs, such that postulated steam line leaks or breaks do not adversely affect the ability to perform safety functions. Safety-related SSCs located inside the safety-related portion of the Reactor Building are protected from the dynamic effects associated with high-pressure steam line breaks by either protective design features (e.g., barriers or blowout panels), are designed for the environmental conditions, are located sufficiently far enough away to avoid the hazards, or a combination of these measures. These design features satisfy the requirements of PDC 4.

Tritium produced in the reactor migrates through the PHTS and IHTS to the steam system. The release of tritium is controlled by collecting liquid discharge to an appropriately sized flash vessel. The flash vessel will evaporate the liquid discharge via a vapor vent to the atmosphere. These design features satisfy the requirements of PDC 60.

The steam system includes radiation monitors on the evaporator, the inlet header prior to the superheater, and the flash vessel vapor vents which monitor tritium releases to the atmosphere during normal operations. These design features satisfy the requirements of PDC 64.

The steam system contains radiological contaminants. Therefore, the design of the system will minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406, as described in Chapter 11.

9.9.1.3 Testing and Inspection

The steam system does not perform any safety-related functions or support safe shutdown of the reactor. The steam system piping and components are periodically inspected for investment protection.

9.9.2 Turbine Generator System

The primary function of the turbine generator system is to convert steam into electricity. One turbine generator building and one turbine generator system are shared between the two reactor units. The turbine generator system interfaces with both the steam system and the condensate and feedwater system. Figure 9.9-1 depicts the turbine generator system.

The commercially available steam turbine generator system converts thermal energy into mechanical energy. The turbine includes rotors, stop valves, control valves, drains, vents, and turning gear. Prior to entering the turbine generator, the superheated steam is attemperated with a portion of saturated steam. The attemperator controls the turbine inlet steam temperature according to turbine manufacturer guidelines.

Turbine stop valves and control valves are the primary methods of steam flow control for normal operations and postulated events. The turbine control valves throttle steam through the turbine in normal operations. A turbine bypass line and associated valve is provided for the superheated steam to directly route to the air-cooled condenser. The turbine bypass steam passes through a desuperheater to lower the temperature prior to entering the air-cooled condenser.

The steam turbine is coupled with the generator. The generator converts the rotational mechanical energy from the turbine into electricity by rotating a magnetic field. The frequency is synchronized between the generator and offsite transmission to transfer electricity to the grid. A portion of the output of the turbine generator is also connected (via a stepdown transformer) to the normal power system and provides power to plant electrical loads, as described in Section 8.2. The turbine control system interfaces with the power generation control system, as discussed in Section 7.2, and provides control of the turbine generator system from the main control room. The turbine generator is equipped with protective monitoring, which will trip the turbine for investment protection in the event of off- normal turbine generator conditions. The turbine generator system is designed to handle a reactor trip on either unit without a corresponding turbine trip. In the event of a reactor trip on one unit, the turbine control system partially closes the turbine control valves to accommodate the reduced steam flow while generating less than 50% electrical load. Residual heat from the tripped reactor is removed