Enclosure 1
Changes to PSAR Chapters 3 and 4
(Non-Proprietary)

Seismic response analysis is performed following Chapter 4 of ASCE 4-16 using deterministic, linear analysis. The relative importance of soil-structure interaction effects, using the characterization of the subsurface materials supporting the SDC-3 structures, defined compatible with those described in Section 2.5, are considered based on the guidance in Chapter 5 of ASCE 4-16.

3.4.1.7 Seismic Qualification

Limit states for SDC-3 SSCs are assigned based on the target seismic performance goals of ASCE 43-19 (see Section 3.4). Specific criteria for the qualification of structures and systems and components are outlined in Section 3.6.

3.4.2 Non-Safety Related SSC Seismic Design (SDC-2 SSCs)

Non-Safety Related Seismic Design (SDC-2) SSCs are designed according to the local building code, the 2012 IBC. For the SDC-2 seismic input, the design basis ground motion is defined in accordance with the deterministic processes of local building code, the 2012 IBC, which refers to ASCE/SEI 7-10 (Reference 6).

Site-specific ground motion parameters are determined per Chapter 21 of ASCE/SEI 7-10. The site response analysis used to inform the SDC-3 (Section 2.5) input will be used to determine the risk-targeted maximum considered earthquake (MCE_R) for the site.

Seismic analysis and qualification of SDC-2 SSCs is also performed in accordance with the 2012 IBC. Seismic design requirements for SDC-2 structures follow Chapter 12 of ASCE/SEI 7-10. Seismic design for SDC-2 systems and components follow Chapter 13 of ASCE/SEI 7-10. Exceptions to ASCE/SEI 7-10 for SDC-2 structures, as required by the Tennessee building code, are applied as needed.

3.4.3 Seismic Instrumentation

Seismic instrumentation that enables the prompt processing of the data at the site is installed for monitoring.

The purpose of the instrumentation is to ~~(1)~~ permit a comparison of measured responses of the site with estimated responses corresponding to the design basis ground motion, ~~(2)~~ and permit facility operators to understand the possible extent of degraded performance within the facility immediately following an earthquake. Instrumentation is also used ~~, and (3) be able~~ to determine when a design-basis earthquake event has occurred that warrants inspection and maintenance activities.

3.4.3.1 Location and Description of Seismic Instrumentation

The seismic instrumentation consists of tri-axial time-history accelerometers located in the free-field and in the safety-related portion of the Reactor Building. The free-field instrument is mounted on rock or competent ground generally representative of the dynamic site characteristics. The instrumentation records time-history data at time increments suitable to capture the range of vibration frequencies in the design basis earthquake spectra. Seismic instrumentation is designed such that if there is a loss of power, recording still occurs. Instrumentation is housed in appropriate weather and creature-proofed enclosures.

3.4.3.2 Seismic Instrumentation Operability and Characteristics

The seismic instrumentation operates during all modes of facility operation. Plant procedures provide for keeping a minimum required number of seismic instruments in service during facility operation. The seismic instrumentation design includes provisions for in-service testing. The seismic instruments are capable of periodic channel checks during normal facility operation and in-place functional testing.

The safety functions of the safety-related portion of the Reactor Building are:

- 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 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.

Table 3.5-1: Load Combinations for the Safety Related Portion of the Reactor Building

Service Level	Load Combination
A	$D + L + T_o + R_o$
B	$D + L + T_o + R_o + E_o$ $D + L + T_i + R_i + E_o$
C	$D + L + T_o + R_o + E_{ss}$ $D + L + T_s + R_s + E_{ss}$
D	$D + L + T_a + R_a + W_t$ $D + L + T_a + R_a + E_{ss}$

Load Nomenclature:

D Dead loads
L Live loads
T_o Thermal loads during startup, normal operating, or shutdown conditions
T_i Thermal loads during Service Level B loadings
T_a Thermal loads during Service Level D loadings
T_s Thermal loads during Service Level C loadings
R_o Pipe reactions during startup, normal operating, or shutdown conditions
R_i Pipe reactions during Service Level B loadings
R_a Pipe reactions during Service Level D loadings
R_s Pipe reactions during Service Level C loadings
E_o Loads generated by 1/3 of design basis earthquake (the design basis earthquake is also the safe shutdown earthquake [SSE])
E_{ss} Loads generated by SSE
W_t Accidental loads due to missile impact effects

3.6.2 Classification of Structures, Systems, and Components

SSCs are assigned safety, seismic, and quality classifications consistent with their safety functions. These classifications are described below. Table 3.6-1 provides a summary of these classifications for all SSCs.

3.6.2.1 Safety Classification

SSCs have two possible safety classifications: safety-related or non-safety related. An SSC is classified as safety-related if it meets the definition of safety-related from 10 CFR 50.2 (with exceptions as described in Section 1.2.3). For the KP-FHR technology, the definition of safety-related is modified from 10 CFR 50.2, to be:

> Safety-related structures, systems, and components means those structures, systems, and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the portions of the reactor coolant boundary relied upon to maintain coolant level above the active core;
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11

Note that for the KP-FHR technology, the definition above reflects an exemption from the definitions in 10 CFR 50.2 that include the terminology "integrity of the reactor coolant pressure boundary." As described in Section 1.2.3 and the Regulatory Analysis for the Kairos Power Salt-Cooled, High Temperature Reactor Topical Report (Reference 1), this exemption is necessary because the technology associated with the KP-FHR is based on a near atmospheric pressure design and the reactor coolant boundary does not provide a similar pressure related or fission product retention function as light-water reactors for which these definitions were based.

SSCs that do not meet the definition, as modified above, are classified as non-safety related.

3.6.2.2 Seismic Classification

SSCs are classified in one of two Seismic Design Categories (SDC) consistent with ASCE 43-19 (Reference 2). Safety-related SSCs are classified as SDC-3. Section 3.4 discusses the SDC-3 classification and Section 3.5 discusses requirements for SSCs that are required to maintain their function in the event of a design basis earthquake. The design basis earthquake is also the safe-shutdown earthquake (SSE). All safety-related SSCs are located in the safety-related portion of the Reactor Building, which is discussed in Section 3.5.1.

The credited safety systems designed to function in a postulated event are described in Chapter 13. For a design basis earthquake, the SDC-3 SSCs that are relied upon to perform a specific credited safety function are listed in Table 3.6-1.

Safety-related systems and components are qualified to maintain their safety function during a design basis earthquake, after a design basis earthquake, or both, depending on the function performed. For example, the reactor vessel is required to perform its safety function (i.e., maintain structural integrity) both during and after a design basis earthquake, whereas the decay heat removal system is required to perform its safety function only after the event, and not during. The specific safety function, therefore, is used to define the ASCE 43-19 Limit State that is used to qualify the SDC-3 SSCs.

Table 4.1-1: Reactor Parameters

Parameter	Value
Thermal Power (MWth)	35
Reactor Coolant Outlet Temperature (°C)	620
Reactor Coolant Inlet Temperature (°C)	550
Reactor Vessel Operating Pressure (bar)	< 2
Reactor Coolant Type	Flibe
Fuel Type	TRISO particle; UCO kernel
Fuel Matrix	Pebble
Equilibrium Fuel Enrichment (wt%)	≤ 19.75
Reflector Type	ET~~U~~-10 Graphite
Control Material	B_4C
Neutron Spectrum	Thermal

coolant level. The design of the reactor vessel allows for online monitoring, in-service inspection, and maintenance.

4.3.1.1.1 Vessel Top Head

The reactor vessel top head (see Figure 4.3-2) is a flat 316H SS disc bolted and flanged to the vessel shell. This interface is designed for leak-tightness but is not credited as being leak tight in safety analyses. The vessel top head controls the radial and circumferential positions of the reflector blocks to ensure a stable core configuration for all conditions (e.g., reactor trip and core motion). The top head contains penetrations, as shown in Figure 4.3-2 and Table 4.3-1, into and out of the vessel and provides for the attachment of supporting equipment and components (e.g., reactivity control elements, pebble handling and storage system components, material sampling port, neutron detectors, thermocouples, etc.). The top head supports the vessel material surveillance system (MSS) which provides a remote means to insert and remove material and fuel test specimens into and from the reactor to support testing.

4.3.1.1.2 Vessel Shell

The reactor vessel is a 316H SS cylindrical shell that, along with the vessel bottom head, serves to form the safety-related reactor coolant boundary within the reactor vessel. It contains and maintains the inventory of reactor coolant inside the vessel. The shell provides the geometry for coolant inlet and vessel surface for the DHRS which transfers heat from the reactor vessel during postulated events. The inside of the shell uses 316H SS tabs to maintain the core barrel in a cylindrical geometry and has a welded connection at the top of the core barrel.

4.3.1.1.3 Vessel Bottom Head

The reactor vessel bottom head is a flat 316H SS disc that is welded to the vessel shell. It contains and maintains the inventory of the reactor coolant inside the vessel, supports the vessel internals, maintains the reactor coolant boundary and provides flow geometry for low pressure reactor coolant inlet to the core. Hydrostatic, seismic and gravity loads on the vessel and vessel internals are transferred to the bottom head and are transferred to the RVSS.

4.3.1.2 Reactor Vessel Internals

The reactor vessel internal structures include the graphite reflector blocks, core barrel and reflector support structure. The vessel internal structures define the flow paths of the fuel and reactor coolant, provide a heat sink, a pathway for instrumentation insertion, control and shutdown element insertion, as well as provide neutron shielding and moderation surrounding the core. The design of the structures support inspection and maintenance activities as well as monitoring of the reactor vessel system.

4.3.1.2.1 Reflector Blocks

The reflector blocks are constructed of grade ET~~U~~-10 graphite. The reflector blocks provide a heat sink for the core and are restrained ensuring alignment of the penetrations to insert and withdraw control elements. The reflector blocks are buoyant in the reactor coolant. The bottom reflector blocks are machined with coolant inlet channels for distribution of coolant inlet flow into the core. The top reflector blocks are machined with coolant outlet channels to direct the coolant exiting from the core into the upper plenum, from which the PSP draws suction. The top reflector blocks also form a pebble defueling chute, as shown in Figure 4.3-1, to direct the pebbles from the core to the pebble extraction machine (PEM), allowing online defueling of the reactor (see Section 9.3). The reflector blocks also provide machined channels for insertion and withdrawal of the reactivity control and shutdown elements described in Section 4.2.2.

Consistent with PDC 35, the reactor vessel internals will assure sufficient core cooling during postulated events and remove residual heat. The safety function of the fluidic diode is to provide a flow path via natural circulation to transfer heat from the reactor core during and following postulated events such that fuel and reactor internal structure damage that could interfere with continued effective core cooling is prevented.

Consistent with PDC 74, the design of the reactor vessel and reflector blocks shall be such that their integrity and geometry are maintained during postulated events to permit sufficient insertion of the control and shutdown elements providing for reactor shutdown.

4.3.3 System Evaluation

The 316H SS structures of the reactor vessel system are fabricated and tested in accordance with Reference 1 standards. The 316H SS vessel internals also satisfy the chemistry restrictions of the ASME Section III code in Division 5, Article HGB-2000. Per the ASME standard, ER16-8-2 weld metal will be used in fabrication of the 316H structures. Commensurate with the safety-related function of the reflector block in ensuring acceptable design limits and maintaining the reactor coolant flow path, quality related controls will be placed on the ET~~U~~-10 graphite. KP-FHR specifications and procurement documents incorporate and reference the applicable guidance and ASME standards. The quality assurance program is described in Section 12.9. These controls demonstrate conformance with PDC 1.

The reactor vessel system makes up a portion of the reactor coolant boundary. The reactor vessel and graphite reflector blocks are therefore designed to maintain geometry during a safe shutdown earthquake to ensure the vessel integrity, insertion of negative reactivity via the RCSS, and to maintain the flow path. The reactor vessel and vessel internals will have dynamic behaviors during a design basis earthquake. These include fluid-structure interaction within the vessel, oscillatory response of components mounted to the reactor top head, i.e., head-mounted oscillators, and relative movement of graphite reflector blocks with respect to one another within the coolant. These dynamic behaviors are accounted for in the design of the reactor and its internals, to ensure continued functionality during and after a design basis earthquake. Models are used to understand fluid migration tendencies considering the pebble bed, reflector blocks, core barrel, and other reactor vessel internal features. The insights gained from the analysis of these models are used to design the reactor to prevent damage to the vessel during a design basis earthquake. The reactor vessel, vessel internals, and vessel attachments such as the RCSS are classified as SDC-3 per ASCE 43-19 "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities" (Reference 2). The reactor vessel will also be protected from the failure of nearby non-safety related SSCs during a design basis earthquake by seismically mounting, physically separating, or using a barrier to preclude adverse interaction, and from failure of attached non-safety related SSCs, such as attached piping (e.g., by design for preferential failure of the non-safety component is a way that does not impact the vessel). These features demonstrate compliance with PDC 2.

The reactor vessel can accommodate internal and external static and dynamic loads. The thermal expansion of the reactor vessel shell and bottom head is supported by the reactor vessel support system (RVSS) (see Section 4.7) during reactor startup, normal operation, and postulated events. Mechanical loadings from static weight, seismic load, and forces from the pebble bed, coolant, and core components are transferred to the vessel shell, to the bottom head, and then to the RVSS. The lateral load path of the vessel support is designed to preclude damage to the decay heat removal system and ensure the vessel maintains its integrity and remains in an upright position. The design of the vessel shell resists hoop stresses from the pressure in the downcomer and supports the transfer of static and dynamic loads between the vessel top head and the vessel bottom head to the RVSS. There are also no pressurized piping systems in or around the reactor vessel, thus precluding high energy line hazards.

Heavy load considerations are addressed in Section 9.8.4, Cranes and Rigging. These features demonstrate compliance with PDC 4.

Core cooling is maintained through the design of the reactor vessel and the reactor vessel internals. As described in Section 4.3.1.2, the vessel and vessel internals define the coolant flow path. To preclude degradation to the vessel due to corrosion of the stainless steel, the reflector blocks and the vessel are “baked” (i.e., heated uniformly) to remove residual moisture prior to coming into contact with coolant. The reflectors, which act as a heat sink in the core, are spaced to prevent the formation of tensile and bending stresses and accommodate thermal expansion and hydraulic forces during normal operation and postulated events. The gaps between the graphite blocks support coolant flow to the reflector thus maintaining a coolable core geometry and precluding reflector degradation by overheating. Maintaining a coolable core geometry and adequate coolant flow through the core ensures the vessel wall temperature is below design limits which prevent vessel failure. Dynamic behavior of the reactor, its support, and its internals are analyzed and designed to ensure vessel integrity and core geometry are maintained in a design basis earthquake to a degree sufficient to ensure passive heat removal. The vessel, as part of the reactor coolant boundary, ensures the containment of radionuclides by ensuring the coolant is confined and the TRISO particles in the fuel pebbles are protected from damage. These features demonstrate conformance to PDC 10.

To demonstrate compliance with PDC 14, the reactor vessel is fabricated, erected, and tested so as to have an extremely low probability of leakage, rapidly propagating failure, and gross rupture. The reactor vessel materials and weld metal will be qualified for use as described in Kairos Power topical report “Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor,” KP-TR-013-P (Reference 3). The 316H SS of the reactor vessel as fabricated and tested in accordance with Reference 1 standards has a high fracture toughness at reactor operating conditions, thus reducing the likelihood of crack propagation. The design of the reactor vessel and vessel internals support a 10-year operating lifetime. This is accomplished by operating the reactor vessel within the as-designed operational and transient condition stresses and by monitoring for changes (e.g., irradiation and thermally induced degradation, corrosion, creep) to the reactor vessel during in-service inspection and testing. The RVSS-reactor vessel bottom head interface is designed to allow access for weld inspections. The reactor vessel top head supports in-service inspection of attachments and penetrations.

The reactor vessel shell and bottom head maintain a coolant pathway for cooling the reactor core and ensure submergence of fuel pebbles in the core. The reactor vessel is fabricated, erected, and tested in accordance with Reference 1 as a Class A component to account for thermal and physical stresses during normal operation and postulated events. The vessel is fabricated from 316H SS base metal and ER16-8-2 weld metal using a gas tungsten arc welding process. Reference 1 provides for weldment stress rupture factors up to a temperature of 650°C for ER16-8-2 weld metal with 316H base metal. Testing provides stress rupture factors up to 816°C for weld material with 316H base metal (Reference 3). The plant control system will detect leakage from the reactor vessel and catch basins are used to detect leaks in nearby coolant-carrying systems. These features demonstrate compliance with PDC 30.

Reactor vessel stress rupture factors are determined up to 816°C to encompass transient conditions. The stress rupture factors are determined by a creep-rupture test on the vessel base material with weld metal under the gas tungsten arc welding process. The vessel precludes material creep, fatigue, thermal, mechanical, and hydraulic stresses. The leak tight design of the reactor vessel head minimizes air ingress into the cover gas and precludes corrosion of the internals. The high temperature, high carbon grade 316H SS of the core barrel and reflector support structure have high creep strength and are resistant to radiation damage, corrosion mechanisms, thermal aging, yielding, and excessive neutron absorption. Load combinations for the reactor vessel system and the RVSS are provided in Table 4.3-2 and Table 4.7-

1. Vessel fluence calculations, as described in Section 4.5, confirm adequate margin relative to the effects of irradiation. The fast neutron fluence received by the reactor vessel from the reactor core and pebble insertion and extraction lines is attenuated by the core barrel, the reflector, and the reactor coolant. Coolant purity design limits are also established in consideration of the effects of chemical attack and fouling of the reactor vessel. These features demonstrate conformance with PDC 31.

The MSS utilizes coupons and component monitoring to confirm that irradiation-affected corrosion is non-existent or manageable. The 316H SS reactor vessel and ER16-8-2 weld material, as a part of the reactor coolant boundary, will be inspected for structural integrity and leak-tightness. As detailed in Reference 3, fracture toughness is sufficiently high in 316H SS under reactor operating conditions that additional tensile or fracture toughness monitoring and testing programs are unnecessary. These features demonstrate conformance to PDC 32.

Fluidic diodes are used to establish a flow path for continuous natural circulation of coolant in the core during postulated events to remove residual heat from the reactor core to the vessel wall. During and following a postulated event, the hot coolant from the core flows from the upper plenum through the low flow resistance direction of the fluidic diode to the cooler downcomer via natural circulation, thereby cooling the core passively. Continuous coolant flow through the reactor core prevents potential damage to the vessel internals due to overheating thereby ensuring the coolable geometry of the core is maintained. The anti-siphon feature also limits the loss of reactor coolant inventory from inside the reactor vessel in the event of a PHTS breach. These features demonstrate compliance with PDC 35.

The downcomer, graphite reflector blocks, and fluidic diodes are passive components designed to maintain structural integrity during postulated events to maintain a natural circulation path and a coolable core geometry for removal of decay heat. The reactor vessel internals are qualified in accordance with Reference 3 and Reference 4 and are designed to perform their function during seismic events as noted above. Based on the design and qualification, there are no credible failure mechanisms within the design basis of the core barrel and the graphite structures that result in a loss of structural integrity. Therefore, degradation of the natural circulation flow path required to support decay heat removal is not expected during normal or postulated events and such failures would be beyond the design basis. However, graphite dust is expected to be present in small quantities in the system and could be postulated to accumulate in portions of the reactor coolant pathway. The functional capability of the normal flow path can be periodically confirmed during operation by monitoring temperature changes to the exit from the reactor vessel. Similarly, the portions of the reactor coolant flow path that are unique to natural circulation (diode pathway and fluidic diode) are capable of being confirmed during normal operations via temperature changes across the diode and across the pathway. Additionally, the fluidic diodes are designed to permit periodic remote inspections via penetrations on the vessel top head to ensure the pathway remains unobstructed. Instrumentation for temperature measurement across the fluidic diode is permitted via the same penetrations used for visual inspection. These features and capabilities demonstrate conformance to PDC 36 and PDC 37. Additional functions performed by the DHRS to support passive decay heat removal are described in Section 6.3.

Highlighted text previously added. Submitted on 8-31-22. (ML22243A250)

The reactor vessel reflector blocks permit insertion of the reactivity control and shutdown elements. The ET~~U~~-10 grade graphite of the reflector blocks is compatible with the reactor coolant chemistry and will not degrade due to mechanical wear, thermal stresses and irradiation impacts during the reflector block lifetime. The graphite reflector material is qualified as described in the Kairos Power topical report "Graphite Material Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor," KP-TR-014 (Reference 4). To preclude damage to the reflector due to entrained moisture in the graphite, the reflector blocks are "baked" (i.e., heated uniformly) prior to coming into contact with coolant and the reactor vessel is design to preclude air ingress. The reflectors, which act as a heat sink in the core,

Table 4.3-2: Load Combinations for the Reactor Vessel System

Service Level	Load Combination
A	$D + L + T_o + P_o + R_o$
B	$D + L + T_o + P_o + R_o + E_o$ $D + L + T_i + P_i + R_i + E_o$
C	$D + L + T_o + P_o + R_o + E_{ss}$ $D + L + T_s + P_s + R_s + E_{ss}$
D	$D + L + T_a + P_a + R_a + W_t$ $D + L + T_a + P_a + R_a + E_{ss}$

Load Nomenclature:

D Dead loads
L Live loads
T_o Thermal loads during startup, normal operating, or shutdown conditions
T_i Thermal loads during Service Level B loadings
T_a Thermal loads during Service Level D loadings
T_s Thermal loads during Service Level C loadings
P_o Pressure loads during startup, normal operating, or shutdown conditions
P_i Pressure loads during Service Level B loadings
P_s Pressure loads during Service Level C loadings
P_a Pressure loads during Service Level D loadings
R_o Pipe reactions during startup, normal operating, or shutdown conditions
R_i Pipe reactions during Service Level B loadings
R_a Pipe reactions during Service Level D loadings
R_s Pipe reactions during Service Level C loadings
E_o Loads generated by 1/3 design basis earthquake (the design basis earthquake is also the safe shutdown earthquake [SSE])
E_{ss} Loads generated by SSE
W_t Accidental loads due to missile impact effects

4.7.3 System Evaluation

The RVSS supports the reactor vessel in the event of an earthquake or other natural phenomenon thus ensuring the integrity of the reactor vessel and its ability to retain reactor coolant. The bottom support meets ASCE 43-19 (2019) (Reference 2) and precludes linear buckling in the vessel support columns under static and design basis earthquake loads. The bottom support is also vertically anchored to the cavity to prevent the vessel from uplift during a design basis earthquake. The vessel connectors meet Reference 2 and provide sufficient lateral and uplift support to the vessel and the vessel top head components. The reactor cavity is also seismically isolated to reduce seismic loads. Load combinations for the RVSS and safety-related portions of the Reactor Building are provided in Table 4.7-1 and Table 3.5-1. These design features demonstrate compliance with PDC 2 for the RVSS.

The RVSS is protected from discharging fluids by catch basins. Sensors and probes installed on catch basins including the bottom support tray can be used as a means of leak detection to preclude damage to the RVSS. There are no pressurized piping systems in proximity to the RVSS thus precluding by design any impacts from high energy line considerations. The RVSS accommodates the reactor vessel temperature loading cycles in combination with relevant mechanical loading cycles to ensure creep-fatigue damages are precluded. The RVSS can also accommodate the growth of the reactor vessel due to thermal expansion between startup and equilibrium conditions. These design features satisfy PDC 4 for the RVSS.

PDC 74 states requires the design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated events (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown. The RVSS maintains the integrity of the reactor vessel by removing heat via the RTMS, actively during normal operation and passively during postulated events. Fission product decay heat and other residual heat from the reactor core is transferred to the reactor vessel; then to the anchored surface by the RVSS. The support columns of the RVSS are sized and spaced to maximize heat transfer between the bottom support and the environment. The thermal break between the RVSS and the reactor building provided by the bottom support insulation ensures the concrete integrity meets ACI 349-13 to support maintenance and inspection of the vessel bottom head/vessel shell weld and to ensure conditions in the surrounding cavity do not exceed maximum allowable parameters. This demonstrates compliance with PDC 74 for the RVSS.

4.7.4 Testing and Inspection

The RVSS temperature will be monitored during operation for conformance with design limits. The RVSS will be included in an in-service inspection program which will be submitted at the time of the Operating License Application.

4.7.5 References

1. ASME Boiler & Pressure Vessel Code, Section III, Division 5 (2019)
2. ASCE 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.”
3. ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary”

Table 4.7-1: Load Combinations for the Reactor Vessel Support System

Service Level	Load Combination
A	$D + L + T_o + R_o$
B	$D + L + T_o + R_o + E_o$ $D + L + T_i + R_i + E_o$
C	$D + L + T_o + R_o + E_{ss}$ $D + L + T_s + R_s + E_{ss}$
D	$D + L + T_a + R_a + W_t$ $D + L + T_a + R_a + E_{ss}$

Load Nomenclature:

D Dead loads
L Live loads
T_o Thermal loads during startup, normal operating, or shutdown conditions
T_i Thermal loads during Service Level B loadings
T_a Thermal loads during Service Level D loadings
T_s Thermal loads during Service Level C loadings
R_o Pipe reactions during startup, normal operating, or shutdown conditions
R_i Pipe reactions during Service Level B loadings
R_a Pipe reactions during Service Level D loadings
R_s Pipe reactions during Service Level C loadings
E_o Loads generated by 1/3 SSE
E_{ss} Loads generated by SSE
W_t Accidental loads due to missile impact effects