**Enclosure 1**
**Changes to Hermes 2 PSAR Chapter 4**
**(Non-Proprietary)**

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-A (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. Table 4.3-3, Table 4.3-4, Table 4.3-5, and Table 4.3-6 provide the qualification tests required for a test reactor with an 11-year lifetime for 316H SS and weld materials. These tables also present (for information only) qualification tests for both a test reactor with a lifetime of 5 years, and a commercial power reactor, which are described in Reference 3. The design of the reactor vessel and vessel internals support an 11-year 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 750°C for weld material with 316H base metal (Reference 3). The plant control system will detect leakage from the reactor vessel with catch basins, as described in Section 4.7, that 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 750°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 design accounts for expected thermal, mechanical, and hydraulic stresses, and precludes failure via material creep, fatigue, and interactions of these phenomena~~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

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.

The reactor vessel reflector blocks permit insertion of the reactivity control and shutdown elements. The ET-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-P-A (Reference 4). Table 4.3-7 and Table 4.3-8 provide the qualification tests required for a test reactor with an 11-year lifetime for graphite materials. These tables also present (for information only) qualification tests for both a test reactor with a lifetime of 5 years, and a commercial power reactor which are described in Reference 4. No additional testing programs, beyond those required for a non-power reactor as described in Reference 4, are anticipated for molten-salt infiltration, oxidation, abrasion, and erosion to support qualifying the graphite material for an 11-year lifetime. 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. The reflectors, which act as a heat sink in the core, are spaced to accommodate thermal expansion and hydraulic forces during normal operation and postulated events. The gaps between the graphite blocks also allow for coolant to provide cooling to the reflector blocks. The reactor vessel permits the insertion of the reactivity control and shutdown elements as well. The vessel is classified as SDC-3 per ASCE 43-19 and will maintain its geometry to ensure the RCSS elements can be inserted during postulated events including a design basis earthquake. These features demonstrate compliance with PDC 74.

### 4.3.4 Testing and Inspection

The reactor vessel and internals will be included in an in-service inspection program, which will be submitted at the time of the Operating License Application.

### 4.3.5 References


1. American Society of Mechanical Engineers, ASME Boiler & Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors." 2017.
2. ASCE 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities."
3. Kairos Power, LLC, "Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor," KP-TR-013-P-A, April 2023.
4. Kairos Power, LLC, "Graphite Material Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor," KP-TR-014-P-A, April 2023.

**Table 4.3-3: Testing Requirements to Extend the ASME Qualification of ER 16-8-2**

| **Test Program** | **Non-power reactor (5-year lifetime)** | **Non-power reactor (11-year lifetime)** | **Commercial power reactor (20-year lifetime)** |
|---|---|---|---|
| Tensile Testing | Test matrix in Table 3 of Reference 3 | No additional testing* | No additional testing* |
| Creep Fatigue Testing | Test matrix in Table 3 of Reference 3 to 760ºC | Test matrix in Table 3 of Reference 3 to 816ºC | No additional testing** |
| Creep Testing*** | Up to 10,000 hours | Up to 20,000 hours | Up to 40,000 hours |

* No additional testing is required beyond that already specified for the non-power reactor (5-year lifetime)
** No additional testing is required beyond that already specified for the non-power reactor (11-year lifetime)
*** See information provided in Section 3.1 of Reference 3

**Table 4.3-4: Testing Requirements for Reactor Design**

| Test Program | Non-power reactor (5-year lifetime) | Non-power reactor (11-year lifetime) | Commercial power reactor (20-year lifetime) |
|---|---|---|---|
| Tensile Testing | Test matrix in Table 4 of Reference 3 | No additional testing* | No additional testing* |
| Stress Relaxation Testing | Test matrix in Table 5 of Reference 3 | No additional testing* | No additional testing* |
| Stress Dip Testing | Test matrix in Table 6 of Reference 3 | No additional testing* | No additional testing* |
| Uniaxial and Notched Bar Creep Testing | Test matrix in Tables 7 and 8 of Reference 3 | No additional testing* | No additional testing* |
| Creep-Fatigue Testing | Test matrix in Table 9 of Reference 3 | No additional testing* | No additional testing* |
| Stress Relaxation Cracking | Test matrix in Table 10 of Reference 3 for Non-Power Test Reactor | Test matrix in Table 10 of Reference 3 for Commercial Power Reactor | No additional testing** |

* No additional testing is required beyond that already specified for the non-power reactor (5-year lifetime)
** No additional testing is required beyond that already specified for the non-power reactor (11-year lifetime)

**Table 4.3-5: Environmental Compatibility Testing of Metallic Materials[1]**

| Test Program | | Non-power reactor (5-year lifetime) | Non-power reactor (11-year lifetime) | Commercial power reactor (20-year lifetime) |
|---|---|---|---|---|
| Corrosion Testing in Nominal Flibe[2] | Effect of Temp. | 600 – 650°C, up to 3,000 hours; 750°C, up to 250 hours | 600 – 650°C, up to 5,000 hours; 750°C, up to 250 hours | 600 – 650°C, up to 10,000 hours; 750°C, up to 250 hours |
| | Welding | 650°C, up to 3,000 hours | 650°C, up to 5,000 hours | 650°C, up to 10,000 hours |
| | Plastic Strain | 650°C, up to 3,000 hours | 650°C, up to 5,000 hours | 650°C, up to 10,000 hours |
| | Aging | 650°C, up to 3,000 hours | 650°C, up to 5,000 hours | 650°C, up to 10,000 hours |
| | Contamination | Air ingress tests with Flibe + air at 650°C | Flibe + int. coolant tests at 650°C | Flibe + int. coolant tests at 650°C |
| | Redox Control | Nominal Flibe + Be, 650°C up to 3,000 hours | Nominal Flibe + Be, 650°C up to 5,000 hours | Nominal Flibe + Be, 650°C up to 10,000 hours |
| | Occluded Geometry | Evaluated using sample and sample cage geometry in all corrosion tests. | | |
| | Erosive Flow | Flibe + graphite particles, 650°C, up to 3,000 hours | Flibe + graphite particles, 650°C, up to 5,000 hours | Flibe + graphite particles, 650°C, up to 10,000 hours |
| | Occlusion of Test System | Assessment of test systems after above testing | | |
| Slow Strain Rate Testing | | Full test matrix on HAZ samples only | Full test matrix on HAZ, Base Metal, and Weld Metal samples | No additional testing** |
| Corrosion Fatigue and Stress Corrosion Cracking | | Full test matrix on HAZ samples only | Full test matrix on HAZ, Base Metal, and Weld Metal samples | No additional testing** |

| Test Program | | Non-power reactor (5-year lifetime) | Non-power reactor (11-year lifetime) | Commercial power reactor (20-year lifetime) |
|---|---|---|---|---|
| Environmental Creep Testing | | 500 – 750°C, testing up to 2,000 hours | No further testing unless environmental degradation observed | No additional testing** |
| Metallurgical Effects | Corrosion | 3,000-hour exposure | 5,000-hour exposure | 10,000-hour exposure |
| | SSRT | $5x10^{-8}$ in/in tests | No additional testing* | No additional testing* |
| | In-Situ Creep | N/A | Flibe + graphite + redox control, 550 – 650°C | Flibe + graphite + redox control, 550 – 650°C |

Notes:

1. Test matrix requirements specified for the 5-year non-power reactor and 20-year power reactor are from Tables 12-17 of Reference 3
2. For corrosion testing in a non-power reactor with an anticipated 11-year lifetime, the total duration of corrosion tests is approximately half of the planned duration relative to the commercial power reactor because the 11-year lifetime of the non-power reactor is approximately half the lifetime of the commercial power reactor.

* No additional testing is required beyond that already specified for the non-power reactor (5-year lifetime)

** No additional testing is required beyond that already specified for the non-power reactor (11-year lifetime)

**Table 4.3-6: Irradiation Effects Testing of Metallic Materials**

| **Test Program** | **Non-power reactor (5-year lifetime)** | **Non-power reactor (11-year lifetime)** | **Commercial power reactor (20-year lifetime)** |
|---|---|---|---|
| Irradiation Induced Embrittlement | Test matrix in Table 11 of Reference 3 | No additional testing* | No additional testing* |
| Irradiation-Affected Corrosion | Test matrix in Table 11 of Reference 3 | N/A, assessed based on inspection and monitoring program | N/A, assessed based on inspection and monitoring program |
| Irradiation-Assisted Stress Corrosion Cracking | Test matrix in Table 11 of Reference 3 | Also assessed via inspection and monitoring program | Also assessed via inspection and monitoring program |

* No additional testing is required beyond that already specified for the non-power reactor (5-year lifetime)

**Table 4.3-7: Qualification Requirements of Unirradiated Graphite Mechanical and Thermal Properties[1]**

| Test Program | Non-power reactor (5-year lifetime) | Non-power reactor (11-year lifetime) | Commercial power reactor (20-year lifetime) |
|---|---|---|---|
| **Mechanical and Thermal Properties** | Based on ASME Sec III Div 5 Code | No additional testing* | No additional testing* |
| **Property Variation** | HHA-III-5000 | No additional testing* | No additional testing* |
| **Purity** | ASTM D7219-08 and ASTM C1233 | No additional testing* | No additional testing* |
| **Molten Salt Infiltration** | Confirmatory testing | No additional testing* | Additional testing only if Flibe infiltration of ET-10 is observed at higher pressure of the commercial power reactor. Additional testing would be then included post-infiltration strength testing. |
| **Oxidation** | ASTM D7542<br>HHA-III-3200 | No additional testing* | No additional testing* |

Note:

1. See information provided in Section 3.0, Table 5 and Table 6 of Reference 4.

* No additional testing is required beyond that already specified for the non-power reactor (5-year lifetime)

**Table 4.3-8 Qualification Requirements for Graphite Irradiation[1]**

| Test Program | Non-power reactor (5-year lifetime) | Non-power reactor (11-year lifetime) | Commercial power reactor (20-year lifetime) |
|---|---|---|---|
| Basic Properties | Use of existing irradiation data for ETU-10 | Use of data within the existing ETU-10 irradiation envelope. If graphite exceeds existing envelope for ETU-10, then new irradiation data will be obtained for ET-10. | Use of data within the existing ETU-10 irradiation envelope. If graphite exceeds existing envelope for ETU-10, then new irradiation data will be obtained for ET-10. |
| Irradiation Creep | Relies on irradiation creep data for other graphite grades | Relies on irradiation creep data for other graphite grades. Based on turnaround estimates from ORNL ETU-10 data and conservative fluence estimates for Hermes 2, 11 years of life is expected to remain pre-turnaround. Uncertainties will be included in the analysis to allow margin. If final design data and turnaround analysis shows that graphite exceeds turnaround fluence, then irradiation creep data for ET-10 will be obtained and used. | Irradiation creep data for ET-10 will be obtained to support 20-year reactor lifetime. |

Note:

1. See information provided in Section 4.3 of Reference 4.