1.3 GENERAL DESCRIPTION OF THE FACILITY

1.3.1 Geographical Location

The facility is located within the East Tennessee Technology Park (ETTP) in Oak Ridge, Tennessee. The facility latitude and longitude are provided in Section 2.1. The site location is illustrated in Figure 2.1-1.

1.3.2 Principal Characteristics of the Site

The site consists of an area located in the northwestern portion of the Heritage Center within the ETTP (the ETTP consists of the Heritage Center, site of former uranium enrichment operations, and the Horizon Center Industrial Park). The property is at the site of the former Buildings K-31 and K-33 of the Oak Ridge Gaseous Diffusion Plant (ORGDP), where uranium enrichment operations occurred from 1954 until the mid-1980s. The overall site is an approximately 185 acre (74.8 hectare) parcel that had been used as farmland prior to the construction of the ORGDP. The site has since been restored to a brown field site by DOE and the former above-grade portions of the buildings were removed.

The site is entirely contained within the ETTP, Oak Ridge, Tennessee. The dominant land use in the site area is a brown field from the ORGDP site. Other operations in the site area are associated with DOE facilities, ongoing conversion of former DOE sites for commercial use, and various industrial activities. Principal characteristics of the site are further described in Chapter 2.

1.3.3 Principal Design Criteria, Operating Characteristics, and Safety Systems

1.3.3.1 Principal Design Criteria

The principal design criteria for the facility are described in Section 3.1. The principal design criteria for the facility are based on the criteria included in Kairos Power Topical Report KP-TR-003-NP-A (Reference 1).

1.3.3.2 Operating Characteristics

The reactor is designed to achieve a reactor power of 35 MWth (design rated thermal power) and a ~~planned operational~~ licensed lifetime of ~~10~~ 4 years. The reactor parameters are provided in Table 4.1-1.

1.3.3.3 Safety Systems

The facility is a fluoride salt-cooled, high temperature reactor. The design of the reactor and fuel are discussed in detail in Chapter 4. The primary heat transport system and primary heat rejection system are addressed in Chapter 5. The safety-system classification is provided in Table 3.6-1.

1.3.4 Engineered Safety Features

Engineered safety features (ESF) are SSCs of the facility designed to mitigate the consequences of postulated events. For the non-power reactor facility, the ESFs are related to the containment of fission products, and the passive removal of decay heat. The ESFs are described in Chapter 6.

1.3.5 Instrumentation, Control, and Electrical Systems

The instrumentation and control (I&C) system monitors and controls plant operations during normal operations and planned transients. The system also monitors and actuates protection systems in the event of unplanned transients. The I&C system is comprised of the plant control system and the reactor protection system. The I&C system is discussed in Chapter 7.

The electrical system provides the normal and backup power to the facility. The electrical system is discussed in Chapter 8.

1.6 SUMMARY OF OPERATIONS

As noted in Section 1.1, the purpose of the non-power reactor facility is to test and demonstrate the key technologies, design features, and safety functions of the KP-FHR technology and its SSCs. The facility will also provide data and insights for the safety analysis tools and computational methodologies used for the design and licensing of a KP-FHR commercial power reactor. The major programs to be performed in the facility will be provided in the application for an Operating License consistent with 10 CFR 50.34(b)(2).

The reactor will be operated for a ~~10~~4-year lifetime over the full range of power to evaluate these aspects of the technology. The process system designs include the necessary features to monitor and assess plant performance in support of these objectives as described elsewhere in this report. The activation product inventory and fission product inventory from the normal operation of the facility and effluent release pathways to the environment, are discussed in Section 11.1 and a description of the radiation sources for the facility will be provided in the application for an Operating License consistent with 10 CFR 50.34(b)(3).

An analysis of postulated events from operation of the facility, including the radiological consequences of unplanned releases, is addressed in Chapter 13.

site boundary where the reactor site management has direct authority over all activities including exclusion or removal of personnel and property from the area.

The EAB is coincident to the site boundary.

The Low Population Zone (LPZ) is 800 meters from the reactor as shown in Figure 2.1-3. The Emergency Planning Zone (EPZ) boundary is set coincident to the site boundary. The EPZ is an area used for emergency activities in the event of an emergency (Reference 6). The doses at the EPZ are below the Environmental Protection Agency (EPA) Protective Action Guide (PAG) Manual guidelines for protective action, as recommended by ANSI/ANS-15.16-2015 (R2020) and pursuant to Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors." This approach is consistent with the allowance for a smaller EPZ in 10 CFR 50, Appendix E.I.3.

2.1.2 Population Distribution

This section provides population distribution data for resident and transient populations for the area within 5 miles (8 km) of the center point of the site for the following years (Reference 9, Reference 10):

- Beginning of the requested license period (2026)
- Five years after the beginning of the requested license period (2031)
- ~~Approximate end of the requested license period (2036)~~

Estimates and projections of resident and transient populations around the site are divided into five distance bands (represented by concentric circles). The distances from the center point of the reactor are: 0 to 0.5 miles (0 to 0.8 km), 0.5 to 1 mile (0.8 to 1.6 km), 1 to 2 miles (1.6 to 3.2 km), 2 to 3 miles (3.2 to 4.8 km), and 3 to 5 miles (4.8 to 8 km). The distance bands are further subdivided into 16 directional sectors, each centered on one of the 16 compass directions and consisting of 22.5 degrees. For each segment formed by the distance bands and directional sectors, the resident population was estimated using the most recent and currently available decennial census year (2010) (Reference 9). The population data is used in the environmental monitoring program discussed in Chapter 11.

2.1.2.1 Resident Population

The distribution of the resident population for the area within 5 miles (8 km) of the site is shown in Figures 2.1-4 to Figure 2.1-7~~8~~. The maps illustrate town, city, and county boundaries.

Figure 2.1-4 shows the population by block group using the most recent and currently available decennial census year (2010) within the site. Figure 2.1-5 also shows the population as of 2010 decennial census but distributes the population into five distance bands based on distance from the center point of the reactor. Population estimates within each quadrant and band were derived from block data, a smaller geographic unit than block groups, also from the 2010 decennial census (Reference 9). To determine the population within each quadrant and band, a population density was calculated for every block within the 5-mile radius. The population was re-calculated based on the area within the quadrants and bands. For each segment formed by the distance bands and directional sectors, the percentage of each block area that falls, either partially or entirely, within that segment was calculated using the geographic information system software known as ArcMap10.5. The equivalent proportion of each block's population was then assigned to that segment. If portions of two or more blocks fall within the same segment, the proportional population estimates for the blocks were summed to obtain the population estimate for that segment, as illustrated in Table 2.1-1.

Figures 2.1-6~~,~~ and 2.1-7~~, and 2.1-8~~ show the population estimates for 2026 (the beginning of the license period)~~;~~ and 2031 (5 years later)~~, and 2036 (approximate end of the license period)~~. Projections are based on county estimates for Roane and Morgan Counties derived from the Boyd Center for Business

and Economic Research, Tennessee's state demographer (Reference 9). The basis of the projection method was the application of a growth rate derived from the county projections to the 2010 decennial census block data. The growth (or loss) rate was determined by calculating the actual yearly percent change of the estimated population growth (or loss) for Roane and Morgan counties as projected by the state demographer. The same annual rates were then applied to the base year of 2010 for each county, which was the most recent decennial census data available, and projected forward for the years 2026~~,~~ and 2031~~, and 2036~~. The same rates were used to project population changes in each distance/direction segment in each county.

Tables 2.1-1 and 2.1-2 show the historical population for 2010 and the projected resident population for the years 2026~~,~~ and 2031~~, and 2036~~ that fall within the distance bands for Roane and Morgan counties (Reference 9, Reference 10).

As shown in Figure 2.1-2, the nearest permanent residence to the reactor is a residence located 0.7 miles away to the northwest. Figure 2.1-3 demonstrates that the nearest resident is outside the LPZ.

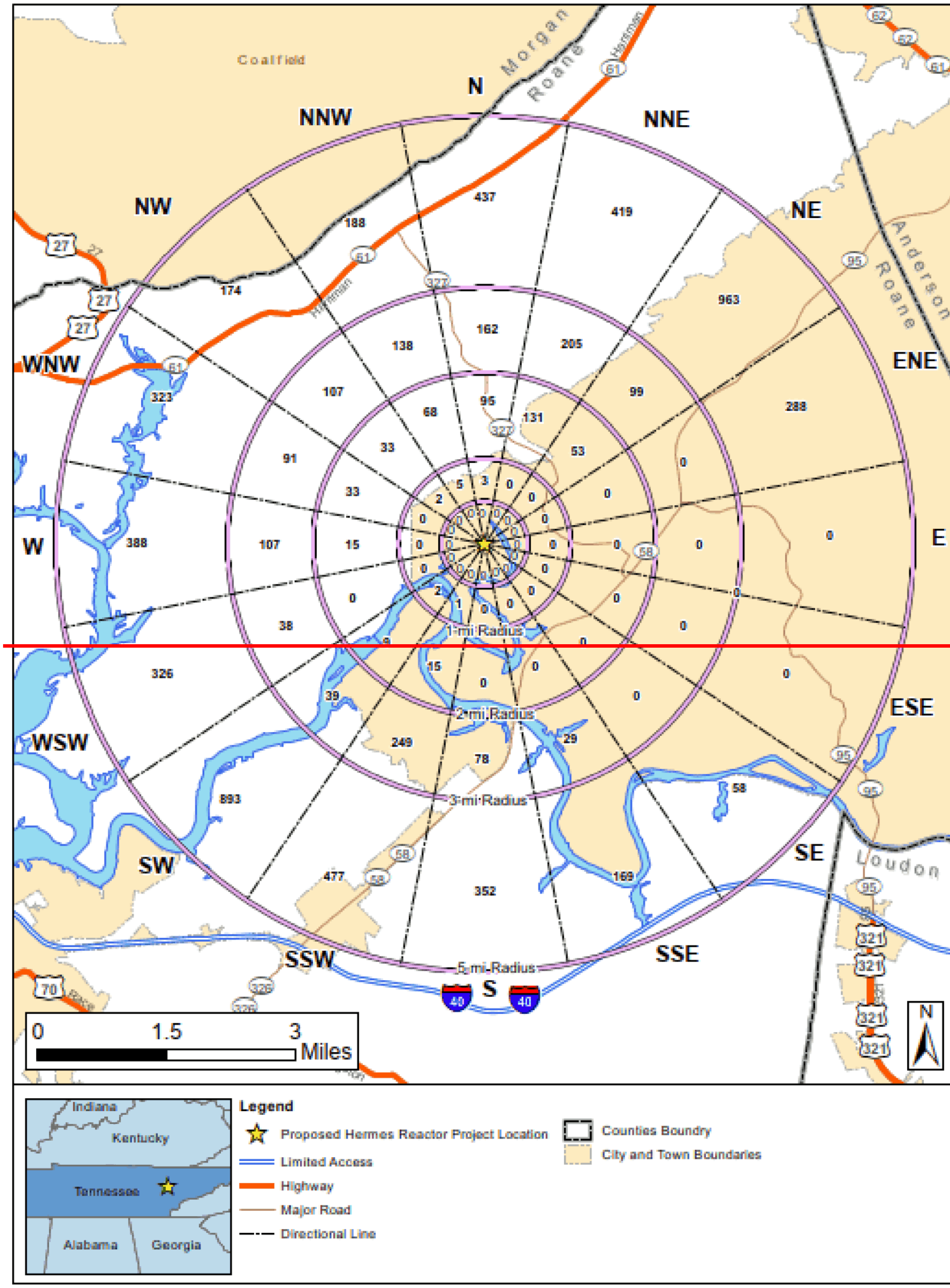
2.1.2.2 Transient Population

Transient populations are temporary or seasonal populations residing in the area, such as in lodging accommodations, dormitories, or classrooms on a college campus. According to the results of the Google Earth desktop research, there are no schools or lodging facilities within 5 miles (8 km) of the site. Thus, there are no transient populations in the area.

2.1.3 References

1. Environmental Systems Research Institute (ESRI), Tennessee Map. 2021.
2. U.S. Department of Energy (DOE) Oak Ridge Environmental Management Program, ETTP fact sheet. 2019. Retrieved from https://www.energy.gov/sites/default/files/2019/01/f58/ETTP%20fact%20sheet_0.pdf.
3. Oak Ridge Office of Environmental Management, East Tennessee Technology Park. Website: https://www.energy.gov/orem/cleanup-sites/east-tennessee-technology-park.
4. Parr, P.D, and Hughes, J.F., Oak Ridge Reservation Physical Characteristics and Natural Resources, Oak Ridge National Laboratory, ORNL/TM-2006/110. September 2006.
5. U.S. Department of Energy (DOE), "Environmental Monitoring Plan for the Oak Ridge Reservation," DOE/ORO--2227/R5. October 2012.
6. ANSI/ANS-15.16-2015(R2020), "Emergency Planning for Research Reactors."
7. U.S. Geological Survey (USGS), "Elevations for Site Buildings," 2021.
8. Not Used.
9. US Census Bureau, 2010 Census—Block Maps. 2010. Retrieved from https://www.census.gov/geographies/reference-maps/2010/geo/2010-census-block-maps.html.
10. Tennessee State Data Center, Boyd Center Population Projections, Population Projections for Tennessee Counties 2019-2070. October 22, 2019. Retrieved from https://tnsdc.utk.edu/estimates-and-projections/boyd-center-population-projections/.

~~Figure 2.1-8: Resident Population Distribution – 2036~~



~~Sources: Reference 1, Reference 9, Reference 10~~

2.3.1.14 Climate Change

While climatic conditions change over time, such changes are cyclical in nature on various time and spatial scales. The timing, magnitude, relative contributions to, and implications of these changes are generally more speculative, even for specific areas or locations. Further, the most extreme projected changes are for time scales much longer than the approximate 4~~10~~-year planned operation period for the Hermes reactor.

Projected changes are generally small compared to natural variation. General predictions of global or United States climatic changes expected during the period of reactor operation are uncertain and are only applicable on a macroclimatic scale. Because the maximum data span available was used in the severe weather analysis, accurate severe weather phenomena have been provided based on best-available historical data. Projections of future severe weather conditions at the site are highly uncertain at best, based on current understanding and modeling of global climate change. Predictions provided by the U.S. Geological Survey (USGS) (Reference 34) vary considerably. For example, one model (the BNU-ESM model) gives a summer maximum temperature increase from approximately 89°F to 93°F with a standard deviation of approximately 3°F over the period of 2025 through 2049.

The Southern Climate Impacts Planning Program is a climate hazards research program whose mission is to help Tennessee residents increase their resiliency and level of preparedness for weather extremes now and in the future. Their research (Reference 35) provides roughly consistent predictions relative to the USGS of average temperature increases between 2010 and 2100 of 4-8°F. This climate prediction also indicates more extreme precipitation events that could have an effect on the threat of flooding potential in general.

The ORR, located in Roane and Anderson Counties in east Tennessee about 25 mi (40 km) west of Knoxville, is managed by the DOE. ORR issues Annual Site Environmental Reports (ASERs), available at https://doeic.science.energy.gov/ASER/. Appendix B of the most recent ASER (for 2019) contains a substantial review of the regional climate for the ORR, including a discussion of climate change trends in Section B.1.

Although the long-term climate trend from multiple sources indicates a moderate increase in the average temperature and possibility of extreme precipitation events, as stated above, through the end of the 21st century, the time scale of the Hermes licensing period is a minor fraction of this projection period.

2.3.2 Local Meteorology

2.3.2.1 Local Meteorological Data Overview

The Hermes Reactor Facility (Reactor Facility) is located at the southeast portion of the site of the former K-33 building of the East Tennessee Technology Park (ETTP) complex. Since the 1940s, this site has been under the jurisdiction of the Atomic Energy Commission (AEC), which became the Department of Energy (DOE) for this function. In the late 1940s, at the request of the AEC, the United States Weather Bureau conducted, for the first time, a meteorological survey of the Oak Ridge, Tennessee, area to provide detailed information regarding wind flow patterns and other factors to determine dispersion of radioactive contaminants (Reference 36). This study led to the establishment of an extensive network of meteorological towers and forecast capability that is still in existence today. A more recent study of the meteorological patterns in the ORNL area was completed in 2011 (Reference 37). The network of meteorological observations provides a strong basis for the onsite meteorological data needed for the site as well as the reactor facility.

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 4~~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. Vessel fluence calculations, as described in Section 4.5, confirm adequate margin relative to the effects

inspection, the pebbles are directed for re-insertion into the core, or to pebble storage for removal from the circulating pebble inventory, based on inspection results.

9.3.1.5 Pebble Inspection

An automated inspection system provides information to the processing portion of the PHSS for determining pebble health. This includes inspection of the physical condition of the pebble for unacceptable wear or damage, identifying moderator and fuel pebbles, as well as an evaluation of the burnup of the fuel relative to a maximum burnup limit using the burn up measurement sensor (BUMS). The burnup measurement is done by means of a gamma spectrometer. Further details pertaining to inspections for wear and damage will be provided with the application for an Operating License.

9.3.1.6 Pebble Insertion

Pebbles are received from the processing system and placed in a buffer storage until required for reinsertion. The pebble buffer storage is sized and orientated to prevent a critical configuration. Individual pebbles are fed into the step feeder insertion machine from this pebble buffer storage as shown in Figure 9.3-2. The pebbles are inserted into the top of the reactor vessel head, then pushed through the insertion line and enter the reactor core via the in-vessel fueling chute at the bottom of the core (see Section 4.3). There is a single active insertion line into the vessel.

9.3.1.7 PHSS Inert Gas Boundary

The components of the PHSS are designed to maintain an inert gas boundary outside of the reactor vessel for pebble handling. The function of the inert gas environment is to prevent absorption of moisture and oxygen into pebbles for pebble handling during normal operations. The inert gas boundary within the PHSS (see Figure 9.3-2) is created by a mechanical structure that encloses the aforementioned components with penetrations for motor shafts, storage outlets, inspection viewport, data channels, electrical power, and pebbles from the off-head conveyance mechanism and for insertion. Portions of the inert gas boundary that are adjacent to personnel access areas have the appropriate radiation shielding.

9.3.1.8 Pebble Storage

Pebble storage is provided for pebble debris, damaged pebbles, spent fuel, and end of life moderator pebbles. The storage portion of the system is composed of a stainless steel storage canister and transporter device. Individual storage canisters are sized to hold approximately 1,900-2,100 pebbles. The dimensions of the canister and quantity of pebbles are sized to maintain a non-critical configuration. A transporter device is used to transfer canisters to either the spent fuel storage area during normal operation or the full core offload area in the event of a periodic maintenance full core offload or an emergent full core offload.

9.3.1.8.1 Spent Fuel Storage

Spent fuel is discharged from service in the core under normal operating conditions, placed in sealed storage canisters, and moved to the spent fuel storage area as shown in Figure 9.3-2. The initial storage area is a cooling pool designed to hold spent fuel canisters while the decay heat of the pebbles drops. The pool is designed to limit radiation exposure to personnel. After cooling in pool storage, the canisters are moved to a concrete storage bay with radiation shielding and forced air cooling. The pool is actively cooled by the CCWS using an in-pool heat exchanger. Water is re-circulated in the pool by the SFCS and make-up water is provided by the treated water system (see Section 9.7.2). The pool and concrete storage bay are designed to prevent a critical configuration. The storage bay ~~is sized~~sizing is sufficient to store spent fuel and moderator pebbles generated during the 4~~10~~ year operating lifetime of the reactor.

13.2 ACCIDENT ANALYSIS AND DETERMINATION OF CONSEQUENCES

13.2.1 Maximum Hypothetical Accident

13.2.1.1 Methodology and Inputs

The calculation of the dose consequences of the MHA uses the source term methods for design basis accidents presented in Reference 1. Section 13.1.1 provides the MHA narrative and assumptions. This section provides a high level summary of the methods used and the inputs to the calculation.

The evaluation of the MHA dose consequences first identifies and accounts for the sources of MAR and the barriers to release. Each barrier is then evaluated for a release fraction to provide dose consequences at the exclusion area and low population zone boundaries.

The four sources of MAR and the associated barriers to release in the MHA:

- TRISO fuel in the reactor core
    - Barriers: TRISO layers, Flibe, and gas space
- Circulating activity
    - Barriers: Flibe and gas space
- Structural MAR
    - Tritium retained by graphite and in Flibe
        - Barriers: Graphite grains (for non-Flibe tritium) and gas space
    - Argon-41 retained in closed graphite pores
        - Barriers: Graphite pores and Gas space

Section 13.1.1 describes several non-physical conditions that are hypothesized to ensure a bounding MHA:

- Pre-transient diffusion of radionuclides from the fuel in the reactor core is neglected: This conservatism is achieved in the evaluation by assuming that the full radionuclide inventory of the fuel is available for release at the initiation of the MHA. The circulating activity is still assumed to be at an upper bound level. Therefore, any MAR originating in the fuel that contributes to the circulating activity is effectively double counted.
- Hypothetical temperature histories are applied to the transient: the hypothetical temperature histories applied to the MHA is provided in Figure 13.2-1. These temperatures set an upper limit for the figure of merit temperatures in the postulated events.
- The gas space is not credited for confinement of the radionuclides that release from the Flibe free surface: radionuclide transport in the gas space barrier is modeled using the conservative building transport and off-site dispersion methods described in Reference 1.
- Conservative, unfiltered, ground level releases: the gas space transport evaluation assumes a conservative leakage rate for the reactor building that releases the entire volume within a 2 hour window to avoid crediting the building as a confinement structure. The dispersion evaluation assumes no radionuclides are filtered after the building transport is evaluated to avoid taking credit for any radionuclide filtering that could occur in the HVAC system.
- Initial tritium inventories are calculated for an assumed 50MWth core that operates ~~at a~~with an assumed 100% capacity factor over ten years. Lower operating powers result in a lower tritium production rate and lower capacity factors allow for the graphite grains to experience time periods of tritium desorption instead of sorption.
- A bounding vessel void fraction of 0.1 is assumed to facilitate the release of low volatility species in the vessel via bubble burst.

15.2 FINANCIAL ABILITY TO OPERATE THE KAIROS POWER FACILITY

Kairos Power expects to apply for a Class 104 license per 10 CFR 50.21(c) (for testing, research, and development activities), and receipt, possession and use of source material under 10 CFR 40, byproduct material under 10 CFR 30, and special nuclear material under 10 CFR 70. Kairos Power financial projections assume a 4~~10~~-year operating period for the non-power reactor facility.

Kairos Power has reasonable assurance of obtaining the necessary funds to cover estimated facility operation costs for the period of the license. Operating costs for the facility will be covered by sustained private investment from Kairos Power investors, with potential supplements from other funding sources. Estimates of the total annual operating costs for each of the first five years of operation of the facility will be provided with the application for an Operating License consistent with 10 CFR 50.33(f)(2).