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Refueling Options and Considerations for Liquid-Salt-Cooled Very High-Temperature Reactors

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ABBREVIATIONS AND ACRONYMS

AGR	advanced gas-cooled reactor
AHTR	Advanced High-Temperature Reactor
AHTR-MI	Advanced High-Temperature Reactor with metallic internals
ANL	Argonne National Laboratory
AVR	Arbeitsgemeinshaft Versuchsreaktor
CCP	core component pot
CLEM	closed loop ex-vessel machine
CRBRP	Clinch River Breeder Reactor Plant
DHX	decay heat exchanger
DRACS	direct reactor auxiliary cooling system
EBR-II	Experimental Breeder Reactor-II
EVST	ex-vessel storage tank
EVTM	ex-vessel transfer machine
FFTF	Fast-Flux Test Facility
FSVR	Fort St. Vrain Reactor
GT-MHR	gas-turbine modular helium reactor
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature reactor
HTTR	High Temperature Test Reactor
IHX	intermediate heat exchanger
IVHM	in-vessel handling machine
IVTM	in-vessel transfer machine
LS-VHTR	liquid-salt-cooled very high-temperature reactor
LWR	light-water reactor
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
ORNL	Oak Ridge National Laboratory
PBR	pebble-bed reactor
PBMR	pebble-bed modular reactor
PCRV	prestressed-concrete reactor vessel
PHX	primary heat exchanger
PRACS	pool reactor auxiliary cooling system
RIM	refueling, inspection, and maintenance
RCB	reactor containment building
RSB	reactor service building
SNF	spent nuclear fuel
THTR	Thorium High-Temperature Reactor

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ABSTRACT

The liquid-salt-cooled very high-temperature reactor (LS-VHTR), also called the Advanced High-Temperature Reactor, is a new reactor concept that combines in a novel way four established technologies: (1) coated-particle graphite-matrix nuclear fuels, (2) Brayton power cycles, (3) passive safety systems and plant designs previously developed for liquid-metal-cooled fast reactors, and (4) low-pressure liquid-salt coolants. The reactor missions include economic production of hydrogen and electricity, with reactor outputs between 2400 and 4000 MW(t) and peak coolant temperatures between 700 and 950°C. The higher temperatures are required for hydrogen production. Several fluoride salt coolants that are being evaluated have melting points between 350 and 500°C, values that imply minimum refueling temperatures between 400 and 550°C. At operating conditions, the liquid salts are transparent and have physical properties similar to those of water.

A series of refueling studies have been initiated to (1) confirm the viability of refueling, (2) define methods for safe rapid refueling, and (3) aid in the selection of the preferred reactor design. This is the first study to examine LS-VHTR refueling; thus, it is a broad examination of refueling challenges and options rather than a detailed study of one specific refueling option. This report examines (1) experience with mechanical operations in molten salt reactors; (2) applicable refueling experience with high-temperature reactors (similar fuel element designs); and (3) applicable refueling experience with sodium-cooled fast reactors (similar plant design with liquid coolant, high temperatures, and low pressures). Three reactor cores with different fuel element designs [prismatic, pebble bed, and stringer (pin-type)] and different refueling strategies are being evaluated. Each is a liquid-salt-cooled variant of a graphite-moderated gas-cooled high-temperature reactor. The findings indicate that several approaches for refueling a LS-VHTR are viable. The study results are described in this report.

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SUMMARY

The liquid-salt-cooled very high-temperature reactor (LS-VHTR), also called the Advanced High-Temperature Reactor (AHTR), is a new reactor concept that combines in a novel way four established technologies: (1) coated-particle graphite-matrix nuclear fuels, (2) Brayton power cycles, (3) passive safety systems and plant designs previously developed for liquid-metal-cooled fast reactors, and (4) low-pressure liquid-salt coolants. The LS-VHTR is the high-temperature variant of the AHTR. Reactor refueling has major impacts on the design and operation of a nuclear power plant. Consequently, a series of refueling studies have been initiated to (1) confirm the viability of refueling, (2) define methods for safe rapid refueling, and (3) aid in the selection of the preferred system design. The study results are described in this report.

No AHTR has yet been built, and no previous studies of refueling such a reactor exist. Therefore, this study examines a wide range of refueling options and concepts. Wherever possible, experience from refueling past or existing reactors is used. There are several major refueling considerations.

- *Fuel type*. As currently envisioned, the LS-VHTR uses graphite-matrix coated-particle fuel. This fuel can be made into several geometric forms (prismatic blocks, pebble beds, and stringers with fuel pins), each with somewhat different refueling requirements. Refueling experience exists for gas-cooled reactors for each of these fuel geometries.

- *Temperature*. Depending upon the specific goals, the peak reactor coolant operating temperatures are between 700 and 950°C. The high-temperature variant of the AHTR is the LS-VHTR for hydrogen production, which will likely operate near 950°C. The AHTR for electricity production may operate at lower temperatures. The candidate liquid-salt coolants have melting points between 350 and 500ºC. If the coolant remains in the reactor vessel during refueling, the salt temperature must be at least 50ºC higher. Hence, a minimum refueling temperature of 550ºC is assumed for options that involve offline refueling. The peak operating temperatures define the temperature requirements for (1) reactor design options that use on-line refueling and (2) refueling and reflector-replacement equipment left inside the reactor during normal operation. Pebble-bed and some stringer fuel options allow online refueling of the reactor.

- *Reactor size*. The expected commercial reactor size is between 2400 and 4000 MW(t), with 18 months between refueling for those design options that do not refuel online. It is required that major fuel-handling operations be accomplished within a few weeks.

- *Pressure*. The reactor operates at atmospheric pressure with coolant boiling points above 1200ºC. Although there is a reactor vessel, there is no heavy-wall pressure vessel that limits access to the reactor core.

- *Environment.* Liquid salts will slowly react with water or moisture in air. An argon or equivalent atmosphere must be maintained above the liquid salt. The liquid salts are transparent and have physical properties similar to those of water.

The LS-VHTR combines technologies from three other reactor technologies (molten-salt reactors, high-temperature reactors, and sodium-cooled reactors); consequently, refueling technology and refueling options are also based on these technologies.

MOLTEN SALT REACTORS

In the 1950s and 1960s, two molten salt reactors (MSRs) were built at Oak Ridge National Laboratory. In these reactors, the fuel is dissolved in the salt and no conventional refueling is required. The MSR salts have the same compositions (except for dissolved fuel and fission products) as the clean salts being considered for the LS-VHTR. However, it was necessary to operate, maintain, and replace components of the MSR (filters, pumps, etc.) and to conduct multiple mechanical operations in the liquid salts at temperatures that exceed those required for refueling of an LS-VHTR.

For a solid-fuel reactor such as the LS-VHTR, refueling is fundamentally a mechanical operation. The several tens of thousands of hours of MSR operating time and the hundreds of thousands of hours of test-loop operating time have developed and demonstrated the technologies required for mechanical operations in liquid-salt environments at high temperatures. This provides design and operating experience in conducting mechanical operations with equipment immersed in liquid salt.

HIGH-TEMPERATURE REACTORS

Three types of fuel geometries have been developed for gas-cooled, graphite-moderated, high-temperature reactors (Fig. S.1). For each fuel type and core design, an equivalent LS-VHTR variant exists. The choice of fuel geometry for the LS-VHTR has not yet been selected; as a result, this refueling study examines (1) the experience base for refueling gas-cooled reactors with each of these three types of fuels and reactor core designs and (2) the applicability of that experience to a liquid-salt-cooled reactor. Each option has specific advantages and disadvantages.

- *Prismatic graphite-block fuel with traditional refueling.* The LS-VHTR would be fueled with prismatic fuel and refueled when shut down. This particular fuel geometry provides a large latitude for the reactor core designer in the choice of (1) fuel-moderator-coolant ratios and (2) core geometry. This option follows the traditional approach to core design used in gas-cooled reactors in the United States. The fuel design that it uses was demonstrated in the Fort St. Vrain reactor in the United States and is used in the operating High-Temperature Test Reactor in Japan. The three-dimensional fuel-assembly structure for traditional batch loading maximizes fuel utilization and design margins when the reactor is operating; however, it also has the most complex refueling operations and has not been demonstrated to be capable of being refueled on line.

- *Pebble bed*. In a pebble-bed LS-VHTR, the core is filled with pebbles, which flow through the core over a period of time. The gas-cooled variant of this option was developed in Germany. A gas-cooled pebble-bed test reactor is operating in China, and a precommercial gas-cooled pebble-bed reactor is currently being built in South Africa. Refueling in a liquid-salt-cooled pebble reactor would be relatively easy and would be done online, as is the case with the gas-cooled reactors. Unlike other fuel forms, the ratio of fuel and moderator to liquid coolant is fixed in a pebble-bed reactor. This places major constraints on the choice of coolant (it most likely will require the use of a salt with enriched lithium-7 and beryllium) and other core design parameters. While this salt is more expensive, it has very low parasitic neutron capture, which combined with the very small excess reactivity and large cylindrical core, would provide high fuel utilization. Initial studies on a liquid-salt-cooled pebble-bed reactor have been conducted at Delft University in the Netherlands.

- *Stringer fuel assembly*. The LS-VHTR can be designed with a graphite core, channels in the reactor core, and stringers of fuel assemblies that fit within holes in the graphite. Conceptually, the core design is similar to the 14 operating British Advanced Gas-Cooled Reactors (AGRs) with the following exceptions: (1) for the LS-VHTR, the gas coolant (carbon dioxide) is replaced with a liquid salt, and (2) a graphite-based fuel assembly may be used, rather than the stainless-steel-clad pins of the AGR. The removal of a single stringer removes all of the fuel in a particular channel in one movement. As with prismatic fuel, stringer fuel allows the coolant void fraction to be controlled, and thus provides the flexibility to use lower-cost salt compositions. As with the AGR, it is potentially possible to perform refueling online with stringer fuel assemblies, allowing similar benefits for capacity factor and fuel utilization as with pebble fuels. Although this design option simplifies refueling, significant challenges remain for an LS-VHTR for this type of fuel in mechanical design, nuclear design, and (perhaps) materials because such a high-temperature fuel assembly has not yet been developed.

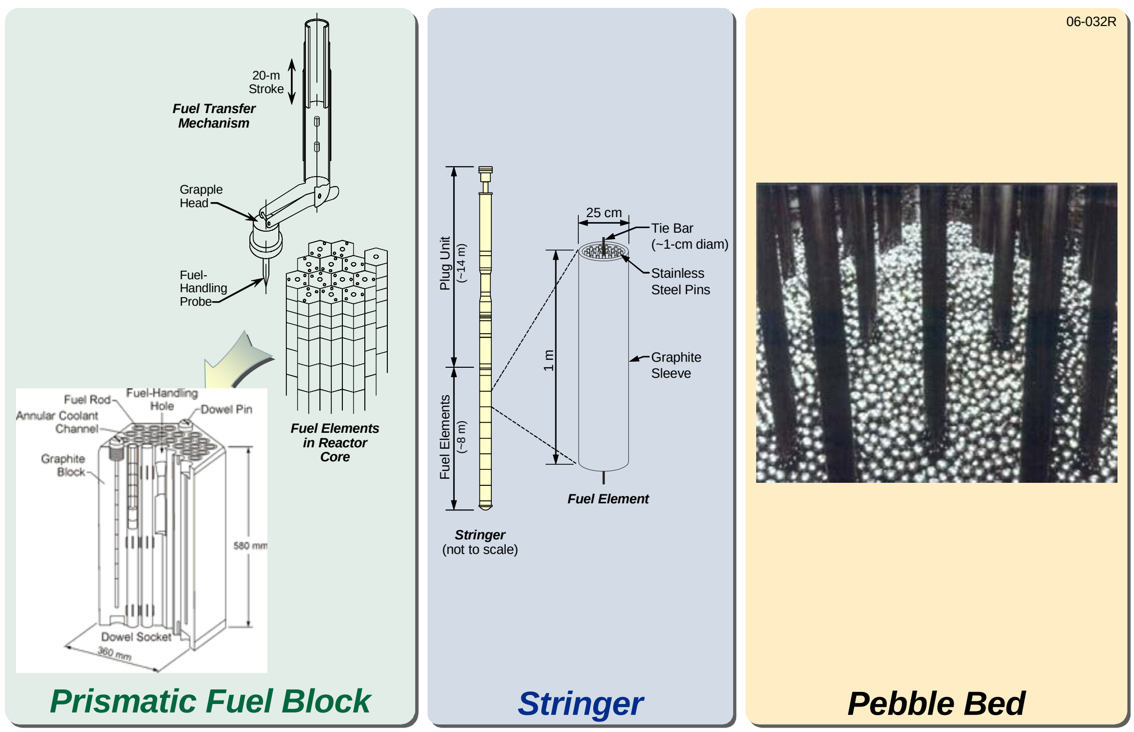

Fig. S.1. High-temperature reactor fuel types and core configurations.

SODIUM-COOLED FAST REACTORS

The LS-VHTR is a low-pressure, high-temperature, liquid-cooled reactor. These attributes are similar to those of a sodium-cooled reactor. Thus, much of the refueling experience base associated with the 20+ sodium-cooled fast reactors that have already been built is directly applicable to the LS-VHTR. The initial LS-VHTR design is based on the General Electric S-PRISM fast reactor. In terms of refueling, sodium-cooled fast reactors have been refueled by leaving the reactor lid on the reactor vessel and refueling through ports in a rotating lid that orientates the refueling machine over the fuel to be replaced. In the United States, the sodium-cooled Experimental Breeder Reactor-II (EBR-II) and the Fast-Flux Test Facility (FFTF) were refueled off-line at near-atmospheric pressure. For the FFTF, the maximum allowable refueling temperature was above 500°C. This refueling experience is directly applicable to the LS-VHTR.

A summary of refueling conditions and experience for various high-temperature reactors is shown in Table S.1. While no previous reactor matches the refueling conditions of the LS-VHTR, a significant experience base in both high-temperature and high-pressure refueling operations exists.

Table S.1. Comparison of refueling operations[a]

		High-temperature reactors					
Environment	**LWR**	**HTGR**	**Pebble**	**AGR**	**LMR**	**MSR**	**LS-VHTR**
Fuel form	Metal-pin assembly	Prismatic graphite	Pebble bed	Metal-pin assembly	Metal-pin assembly	Liquid	Prismatic graphite
Temperature (ºC)	<<100	<100	950	640 (750)	370 (540)	654	550
Pressure (bar)	1	1	40	43	1	1	1
Coolant environment	Water	Helium	Helium	Carbon dioxide	Sodium	Molten salt	Liquid salt
Visibility	Clear	Clear	Clear	Clear	Opaque	Partly opaque	Clear
Assemblies per MW(e)	0.2−0.4	4.5		0.5	0.5	0	0.2−2
No. of reactors	>300	2	3	14	>20	2	0
Timing	Off-line	Off-line	Online	Online	Off-line	Online	Off-line

[a]All data from actual operations, except for the LS-VHTR. Nominal and peak (in parenthesis) allowable temperatures during fuel-handling operations are shown. LWR = light-water reactor; HTGR = high-temperature gas-cooled reactor; AGR = advanced gas-cooled reactor; LMR = liquid-metal reactor; MSR = molten salt reactor; AHTR = Advanced High-Temperature Reactor.

Instrumentation is required to support refueling, inspection, and maintenance (RIM) operations. An examination of the options was conducted—including new optical instrumentation options made possible by the transparent characteristics of the coolant. Relatively new optical techniques, such as laser range finding and creation of three-dimensional images, may enable much better viewing and control of refueling and other operations than have traditionally been possible in sodium-cooled or gas-cooled reactors.

The refueling studies that have been initiated have led to several conclusions.

- *Limits*. No fundamental difficulties were identified for refueling an LS-VHTR. In terms of refueling conditions (temperature, salt environment, pressure, etc.), the key technologies and experience for refueling operations already exist. However, no refueling has been performed under the specific combination of conditions that is expected for the LS-VHTR. As with any new reactor concept, the development of the refueling machine will require significant detailed mechanical design and testing.

- *Fuel forms*. The three alternative fuel forms for an LS-VHTR have very different characteristics and different refueling strategies. Trade studies will be required to determine the optimum reactor core and fuel configuration. Our preliminary assessment suggests that both pebble bed and stringer-type cores would be easier and faster to refuel than the Fort St. Vrain prismatic cores.

- *Instrumentatio*n. The optical characteristics of the liquid-salt coolant and recent advances in optical technologies may result in major improvements in instrumentation capabilities relative to those used in existing sodium-cooled and high-temperature reactors.

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1. INTRODUCTION

The Advanced High-Temperature Reactor (AHTR) is a new reactor concept (Ingersoll and Forsberg, 2006) that uses a liquid-salt coolant and a solid fuel. The liquid-salt-cooled very high-temperature reactor (LS-VHTR) is the high-temperature variant of the AHTR designed primarily for hydrogen production with peak coolant temperatures near 950°C. A critical economic issue for all reactors is safe fast refueling for high plant availability. The liquid salts have high melting points; thus, the AHTR refueling temperatures will be between 400 and 550°C, higher than those for many other reactors. Furthermore, the refueling operations will be conducted in liquid salt. A series of investigations were initiated to understand the refueling challenges and options. Because no LS-VHTR has yet been built, there is no directly applicable experience in these activities. However, a large experience base exists in refueling many types of high-temperature reactors (Paget, 1967). Integrated studies at Oak Ridge National Laboratory (ORNL), Areva NP, the University of California at Berkeley, and Argonne National Laboratory (ANL) were initiated to address these issues.

Section 2 provides a description of the LS-VHTR and identifies specific issues associated with refueling.

Refueling any solid-fuel reactor involves primarily mechanical operations to replace spent nuclear fuel (SNF) from a reactor core. While no direct experience with solid-fuel refueling operations in a salt coolant exists, several molten salt reactors (MSRs) were built in which the fuel is dissolved in the coolant. Although there were no solid fuel assemblies and thus no traditional refueling operations associated with these MSRs, many types of mechanical operations were conducted with the equipment immersed in liquid salt that demonstrate mechanical operations in this environment. Section 3 summarizes this experience base.

As currently envisioned, the LS-VHTR uses a graphite-matrix coated-particle fuel. Three major types (prismatic, pebble bed, and assembly) of fuel can be fabricated. Each has different refueling demands. Section 4 discusses the alternative fuel geometries and the implications for refueling and core design.

Section 5 reviews the applicability of sodium fast reactor refueling to an LS-VHTR. Both the LS-VHTR and sodium-cooled fast reactor can be described as high-temperature, low-pressure, liquid-cooled reactors that require control of the chemical composition of the gas space above the liquid. Because of the functionally similar characteristics of these two reactor classes, many of the technical characteristics associated with refueling a sodium fast reactor are directly applicable to a LS-VHTR.

Refueling, inspection, and maintenance (RIM) operations require special instrumentation. While the LS-VHTR will have the traditional high-temperature reactor instrumentation, the optically clear characteristics of the salt create new instrumentation possibilities. These are described in Sect. 6.

Lastly, Section 7 summarizes some of the conclusions derived from this study.

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2. DESCRIPTION OF THE LS-VHTR

2.1 SYSTEM DESCRIPTION

The LS-VHTR, also called the AHTR, is a new reactor concept that combines in a novel way four established technologies: (1) coated-particle graphite-matrix nuclear fuels, (2) Brayton power cycles, (3) passive safety systems and plant designs previously developed for liquid-metal-cooled fast reactors, and (4) low-pressure liquid-salt coolants. The peak coolant temperatures are between 700 and 950°C, with the specific design temperature dependent upon the mission. The highest-temperature variant of the AHTR, the LS-VHTR, will be for hydrogen production and may have peak coolant temperatures of ~950°C. The AHTR designs for electricity production will likely have lower peak coolant temperatures.

A system schematic is shown in Fig. 2.1, while Fig. 2.2 shows a specific design concept. During operation, heat is transferred from the reactor core by the primary liquid-salt coolant to an intermediate heat-transfer loop. The intermediate heat-transfer loop uses a secondary liquid-salt coolant to move the heat to a thermochemical hydrogen production facility or to a turbine hall to produce electricity. If electricity is produced, a multi-reheat nitrogen or helium Brayton power cycle (with or without a bottoming steam cycle) is used. Electrical efficiencies are expected to be near 50%. Alternative design configurations are being considered, including three alternative-decay-heat removal systems, five alternative fluoride salts, and three reactor fuel configurations.

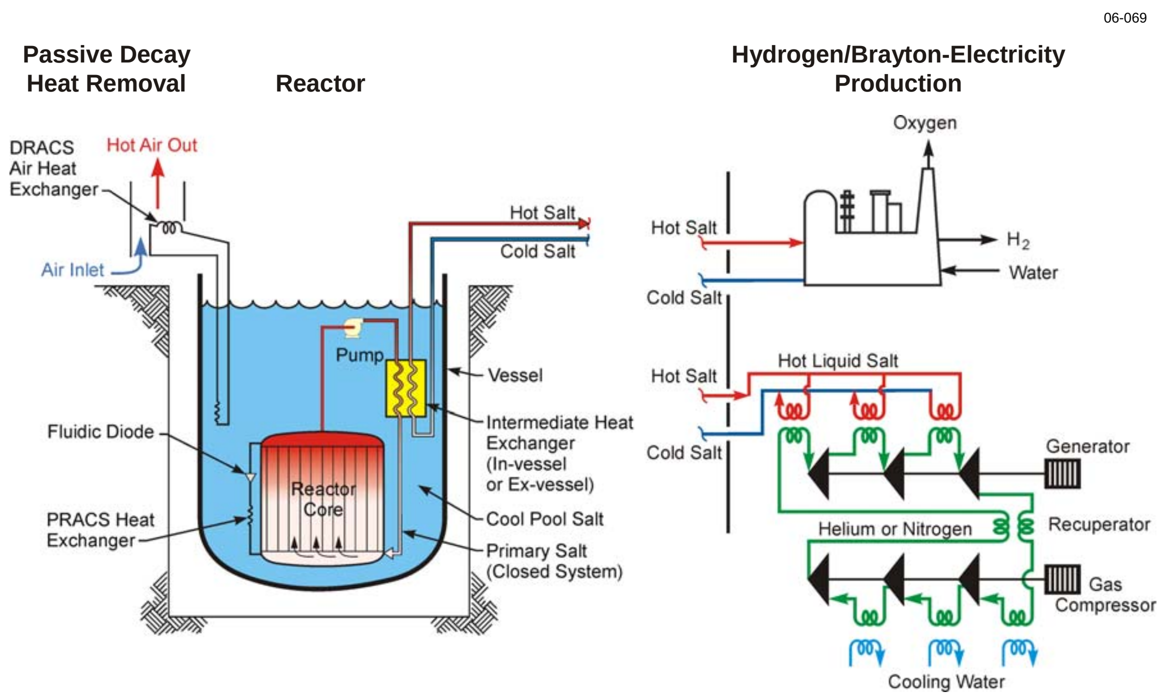

Fig. 2.1. Schematic of the LS-VHTR. (DRACS = direct reactor auxiliary cooling system; PRACS = pool reactor auxiliary cooling system.)

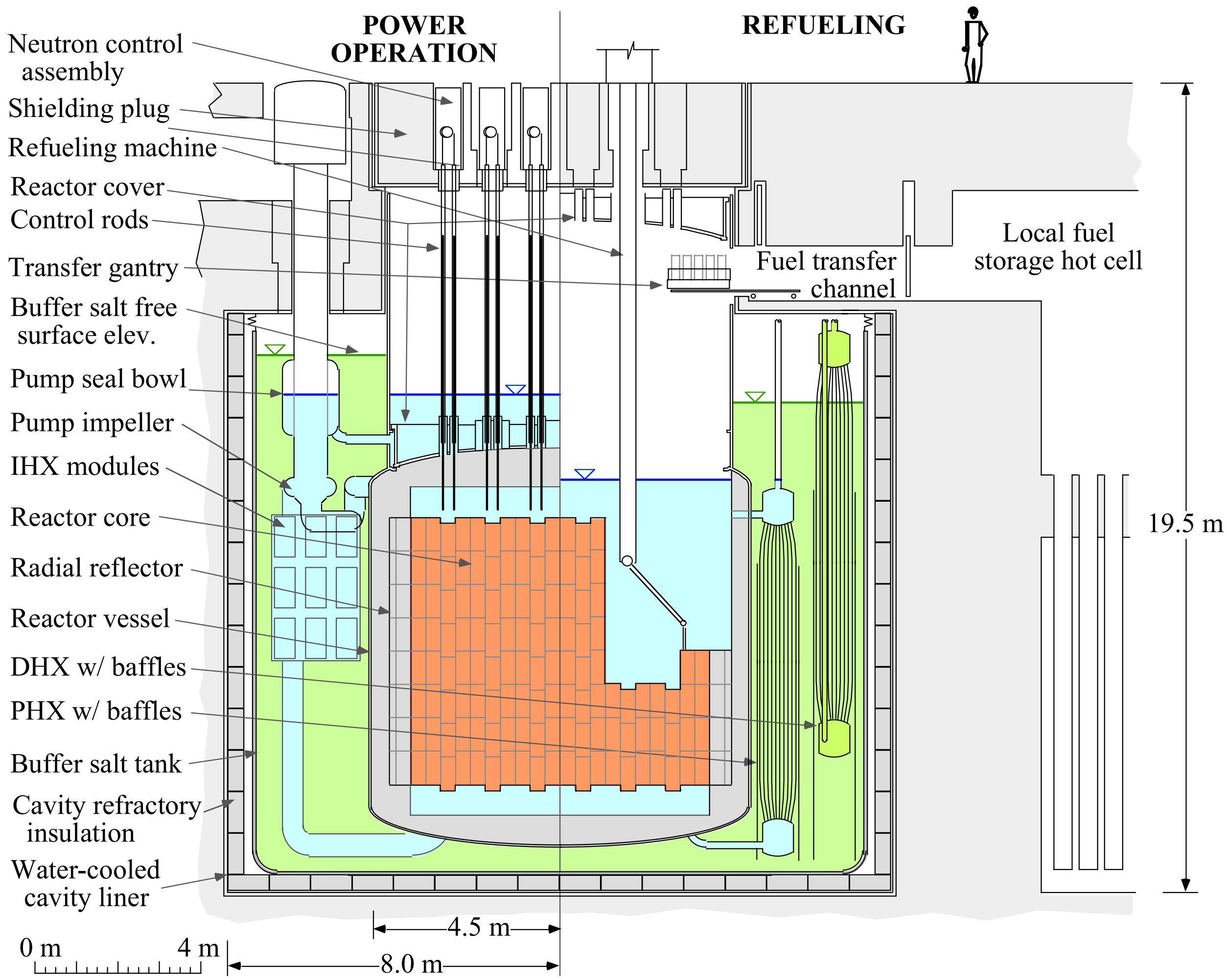

Fig. 2.2. Elevation view of the AHTR with metallic internals (AHTR-MI) for normal operation (left) and refueling (right) modes. IHX = intermediate heat exchanger; DHX = decay heat exchanger; PHX = pool heat exchanger (*Source: Peterson and Zhao, 2006.*)

The LS-VHTR uses the same type of coated-particle graphite-matrix fuel that has been successfully used in high-temperature gas-cooled reactors such as the Peach Bottom Reactor, the Fort St. Vrain Reactor (FSVR), the Arbeitsgemeinshaft Versuchsreaktor (AVR), and the Thorium High-Temperature Reactor (THTR). At this time, graphite-based fuels have been demonstrated to be compatible with only two coolants: helium and fluoride salts.

The optically transparent liquid-salt coolant is a mixture of fluoride salts with freezing points near 400°C and atmospheric boiling points of ~1400°C. The reactor operates at near-atmospheric pressure, and at operating conditions, the liquid-salt heat-transfer properties are similar to those of water. Several different salts can be used as the primary coolant, including lithium-beryllium, sodium-beryllium, and sodium-zirconium fluoride salts.

The baseline 2400-MW(t) reactor layout has strong similarities to those for sodium-cooled fast reactors and has subsystems similar to those in the Experimental Breeder Reactor-II (EBR-II) and the S-PRISM sodium-cooled 1000-MW(t) fast reactor designed by General Electric. These similarities are a direct consequence of the fact that both the LS-VHTR and sodium-cooled reactors are high-temperature liquid-cooled reactors that operate at near-atmospheric pressure. The baseline LS-VHTR uses a passive direct reactor auxiliary cooling system (DRACS) similar to that developed for EBR-II and the European Fast Reactor (EFR). Alternative decay-heat cooling systems are being evaluated (Forsberg, 2006a).

Initial economic assessments (Forsberg, 2006b) indicate capital costs per kilowatt (electric) that are 50 to 55% of those for modular ~600 MW(t) high-temperature gas-cooled reactors and the ~1000 MW(t) S-PRISM reactor—primarily because the high volumetric heat capacity and natural circulation characteristics of the liquid salt allow the AHTR to achieve passive safety at much higher power output from a similar size reactor. Other assessments indicate significantly lower overnight capital costs relative to those of advanced pressurized water reactors because of (1) the higher efficiency, (2) the use of a lower-cost Brayton power cycle to replace the steam Rankine cycle, and (3) the use of smaller primary-loop equipment due to the properties of the liquid salt coolant.

2.2 COOLANT SALT OPTIONS AND IMPLICATIONS FOR REFUELING

Five closely related fluoride salts that have similar properties are being evaluated as coolants (Table 2.1), each with specific advantages and disadvantages. Studies are under way to determine the optimum fluoride salt (Williams, 2006). The salt properties, particularly the melting point and the density, can have major impacts on refueling. The high melting points relative to other coolants have a major impact on the refueling technology.

Table 2.1. Candidate liquid salts

Fluoride Salt	Composition (mole %)	Melting point (ºC)	Density (g/cm^3)	Notes
^{7}LiBe	67–33	460	1.94	Best neutronics, Be toxicity, high Li cost, experience: MSRE, no gamma emitters, low density (fuel buoyancy)
NaBe	57–43	340	2.01	Low MP, Be toxicity, low density (fuel buoyancy)
^{7}LiNaZr	26–37–37	436	2.79	Small addition of expensive Li (2 wt %), low toxicity
NaZr	59.5–40.5	500	3.14	Inexpensive, low toxicity, experience: aircraft nuclear propulsion salt
NaRbZr	33–23.5–43.5	420		Inexpensive, max freeze protection with minimum Rb, max Zr without high ZrF_4 vapor pressure, neutronic questions

[a]MSRE = Molten Salt Reactor Experiment

The high temperatures will require special considerations in the design of the refueling machines and will likely require that the fresh fuel be preheated before refueling. These features exist in refueling machines for sodium-cooled fast reactors such as the Fast Flux Test Facility (Sect. 5 and Appendix A) and in refueling machines for the British Advanced Gas-Cooled Reactors (Sect. 4.3).

Depending upon the choice of salt and fuel, the fuel may be more or less dense than the salt (i.e., the fuel may float in the salt, a phenomenon that occurs in some other reactor systems). In most reactor systems the fuel is held down because hydraulic forces during normal or off-normal conditions can cause the fuel to move. However, for refueling operations the hold-down mechanisms must be released. The relative densities of the fuel and coolant must be accounted for. The implications of these factors for refueling are discussed in Sect. 4—including options such as adding ballast to the fuel assembly to achieve negative buoyancy or using hold-down mechanisms.

The fuel density is dependent upon the details of the fuel design and the fabrication process. Most, but not all, graphite-matrix coated-particle fuels have densities between 1.7 and 1.8 g/cm^3, values that are significantly lower than the theoretical densities of the materials. This reflects both design and manufacturing requirements. An example of a design requirement in the fuel microspheres is for a low-density layer of carbon that (1) stores fission product gases and (2) allows thermal expansion or contraction of the harder silicon carbide layer without failure. Manufacturing systems result in graphite densities that are significantly less than the theoretical density of graphite. The density can be increased; however, this process involves significant added costs with few benefits.

Liquid salts do not wet the graphite; thus, the SNF assemblies are expected to drain clean of salt during refueling operations. Limited studies have been done on the disposal of this fuel in repositories (Forsberg 2006c); however, no studies have yet been done on whether additional cleaning of the SNF will be required before long-term storage or disposal to remove any residues.

2.3 FUEL DESIGNS AND IMPLICATIONS FOR REFUELING

Three alternative fuel designs (prismatic, pebble bed, and stringer-assembly) are being considered. As discussed in Sect. 4, each fuel design requires a different refueling strategy. Historically in gas-cooled reactors, prismatic fuels use off-line refueling while pebble and stringer refueling can be performed online.

In last 40 years, the fuel burnup in light-water reactors (LWRs) has increased from ~20,000 to ~60,000 MWd/ton. This increased burnup has reduced refueling operations and was a major factor in reducing refueling times. The same economic drivers exist for the LS-VHTR and may result in significant reductions in refueling time as the technology is developed.

Coated particle fuels have been shown to be capable of sustaining much higher burn up levels than LWR fuels. Refueling durations for reactors with offline refueling will ultimately be constrained by the limitation to keep initial enrichment levels below 20% ^{235}U. Preliminary analysis suggests that the prismatic fuel LS-VHTR can achieve 18-month refueling intervals, as in current LWRs. Pebble and stringer fuel LS-VHTRs may be refueled online and will not require refueling outages. However, periodic graphite reflector replacement will be required but at much longer intervals than required for LWR refueling.

2.4 DECAY HEAT AND IMPLICATIONS ON REFUELING

Reactors that are refueled offline are refueled shortly after reactor shutdown which means that the decay heat levels are very high. In many reactors, the decay heat from the SNF is a major design and operational constraint in refueling. In sodium-cooled fast reactors, it is usually the primary design constraint in terms of refueling. For the LS-VHTR, however, this major refueling challenge will be significantly less than in water-cooled and sodium-cooled power reactors for the following reasons.

- *Power density*. The expected power density of the LS-VHTR is about 10 W/cm^3 vs 50 W/cm^3 for boiling-water reactors, 100 W/cm^3 for pressurized water reactors, and several hundred watts per cubic centimeter for sodium-cooled fast reactors. The decay heat from SNFs is roughly in proportion to the power densities during operations.
- *Peak fuel temperatures*. The high-temperature reactor fuels operate at temperatures up to 1250°C. This factor makes it much more difficult to cause fuel failure during refueling by overheating than is the case with other types of SNF.

To support a scoping analysis of LS-VHTR refueling system designs, adiabatic temperature rise per unit time in a prismatic fuel element during the core refueling was calculated (Cahalan and Taiwo, 2006) and is shown in Fig. 2.3. These calculations provide input for determination of in-vessel post-shutdown cool-down times prior to refueling, as well as selection of ex-vessel cooling requirements during SNF transport from the reactor to the SNF storage facility. The decay heat curves generated for the LS-VHTR core were generated by Kim, Taiwo, and Yang (2005) based on a design by Ingersoll et al. (2005). The reference LS-VHTR design has a power density of 10 MW/m^3. For this study, power densities bounding this value have been considered, because there would be a distribution in core power densities for fuel elements at operating conditions. For this reason, estimates of temperature rises have been made for cases with fuel-element initial power densities of 6.6, 10, and 15 MW/m^3. Results are presented in Fig. 2.3 for the range of 1 to 50 days.

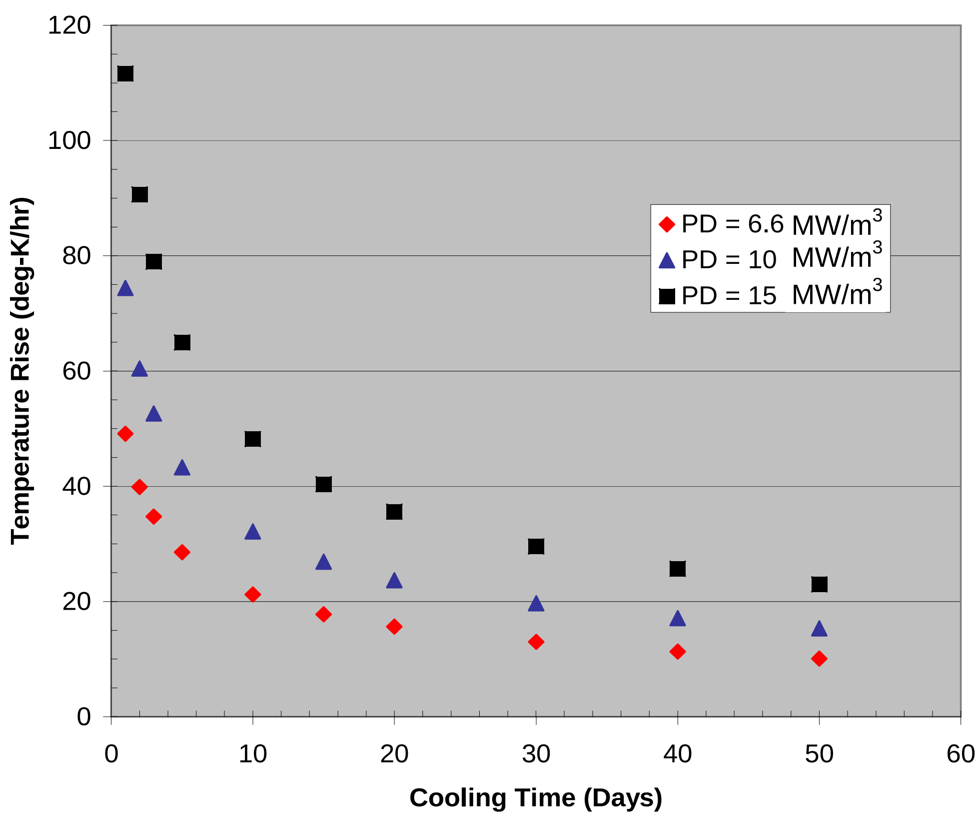

Fig. 2.3. Temperature rise per unit time as a function of cooling time. PD = power density.

A temperature rise greater than 500 K/h was observed immediately after shutdown in the three cases. For pebble and stringer fuels that will be refueled online, reliable cooling of the removed SNF will be required as is done for gas-cooled high-temperature reactors with on-line refueling (Sect. 4.3). For offline refueling, there will be a day or two after shutdown before the reactor refueling is started. This greatly reduces the cooling requirements for the SNF, as shown in Fig. 2.3, and in most cases will allow conduction and natural circulation to provide the required decay-heat cooling—as has been done in the refueling of high-temperature gas-cooled reactors (Sect. 4.1). The estimates in Fig. 2.3 provide a conservative upper bound for the type of temperature rises that would be obtained.

3. MECHANICAL OPERATIONS IN LIQUID SALT: EXPERIENCE FROM MOLTEN SALT REACTORS

In an MSR, the fuel is dissolved in the salt. Between the mid 1950s and the early 1970s, the billion-dollar MSR programs in the United States built two reactors at ORNL (*Nuclear Applications and Technology* 1970; Forsberg, 2006d). These programs included post-operation examination of components (Thoma 1971) that provide a starting point to understand longer-term behavior. The Aircraft Reactor Experiment, a 2.5-MW(t) reactor that operated in 1954 at a peak temperature of 882°C, was part of a large effort to develop an aircraft nuclear propulsion system. This was followed by the Molten Salt Reactor Experiment (MSRE), a highly successful 8-MW(t) reactor that operated for several years at peak temperatures of 654ºC and demonstrated most of the key technologies required for a power reactor. This included 21,788 h of pump operating time and 17,655 h with energy production. The MSRE was part of an effort to develop a commercial breeder reactor that used the ^{233}U/thorium cycle. In the corresponding development programs, there were hundreds of thousands of hours of operating time on a variety of high-temperature salt loops. Detailed plans for a large commercial reactor were developed. Table 3.1 (Rosenthal 1972) lists the pump tests associated with these programs, describes the various pumps, and gives some perspective on the operational experience that was gained.

Table 3.1. Characteristics and operation time for ORNL salt and liquid metal pumps

Model	Fluid[a]	Head (ft)	Flow (gal/min)	Speed (rpm)	Temperature (°F)	Number built	Total hours
LFB	Na, NaK, and MS	92	5	6000	1100–1400	46	466,000
DANA	Na, NaK, and MS	300	150	3750	1000–1500	10	57,000
DAC	MS	50	60	1450	1000–1400	3	4,000
In-Pile Loop	MS	10	1	3000		8	14,000
MF	NaK and MS	50	700	3000	1100–1500	3	41,000
PKA	NaK and MS	400	375	3550	700–1500	2	21,500
PKP	NaK and MS	380	1500	3500	700–1500	4	45,000
MSRE fuel salt pump	MS and Helium	50	1200	1175	1000–1225 100–1200	2	31,600 6,000
MSRE coolant salt pump	MS and Helium	78	800	1775	1000–1225 100–1200	2	24,600 4,000
MSRE Mark-2 fuel salt pump	MS	50	1200	1175	1000–1300	1	14,000
ALPHA	MS	300	30	6500	850–1400	1	6,000
Total						**82**	**734,700**

[a]MS refers to tests in liquid salts (clean salts) and molten salts that contained uranium, thorium, and fission products.

These large programs developed the base technology for handling liquid fluoride salts including pumps, bearings (Smith 1961), valves, filters, cover-gas isolation systems between the salt and the atmosphere, and a wide variety of other mechanical equipment. This initiative also included extensive work with graphite. The MSR uses graphite in the reactor core as a moderator. Key results of these programs included (1) the excellent compatibility of graphite with liquid salts, (2) the viability of handling liquid salts in nuclear reactors, including keeping the salts in a liquid state under a wide variety of normal and shutdown conditions, (3) the development of compatible materials of construction up to 750ºC, and (4) the stability of liquid salts in intense radiation fields. This experience base provides engineering confidence that mechanical operations can be conducted in nuclear-reactor liquid-salt environments at temperatures substantially above the refueling temperature of an LS-VHTR.

Corrosion testing was conducted for a wide variety of materials in both clean salt systems and molten salt systems (liquid salts with high concentrations of dissolved uranium and fission products). Nuclear code-qualified materials of construction were developed for liquid salts up to temperatures of 750°C. Corrosion rates in clean salt were very low compared with those in uranium-bearing salts and compared to corrosion rates seen in other power reactors. The ORNL experience provides real-world data on operations with liquid salts that have compositions similar to those being considered for the LS-VHTR.

Because it is a liquid-fuel reactor, the MSR did not require traditional refueling equipment. The fuel was simply drained or pumped from the reactor. However, the graphite moderator in the reactor core was expected to be damaged by high-energy neutrons and to require replacement one or more times in the lifetime of a commercial reactor. Although studies were conducted to determine how to replace the graphite, both by removal as a single unit attached to the reactor cover and removal as individual moderator assemblies, these processes were not demonstrated. For pebble bed and stringer fuels, which are refueled online, similar methods for replacement of graphite reflectors will be required. For prismatic fuel, the reflector elements will be replaced using similar methods as used to replace the prismatic fuel elements.

4. HIGH-TEMPERATURE REACTOR REFUELING EXPERIENCE

Three types of gas-cooled, graphite-moderated, high-temperature reactors have been built: (1) high-temperature gas-cooled reactors (HTGRs) with prismatic fuel elements, (2) pebble-bed reactors (PBRs), and (3) advanced gas-cooled reactors (AGRs) with stringer fuel assemblies. The fuel forms are shown in Fig. 4.1. Each reactor type has a different fuel geometry and uses a different approach for refueling. A corresponding LS-VHTR variant exists for each reactor and fuel geometry using graphite-matrix coated-particle fuel. Descriptions of the refueling systems are provided for the three-demonstrated gas-cooled reactor types that have proven refueling systems, along with observations about the equivalent LS-VHTR variants that incorporate salt cooling.

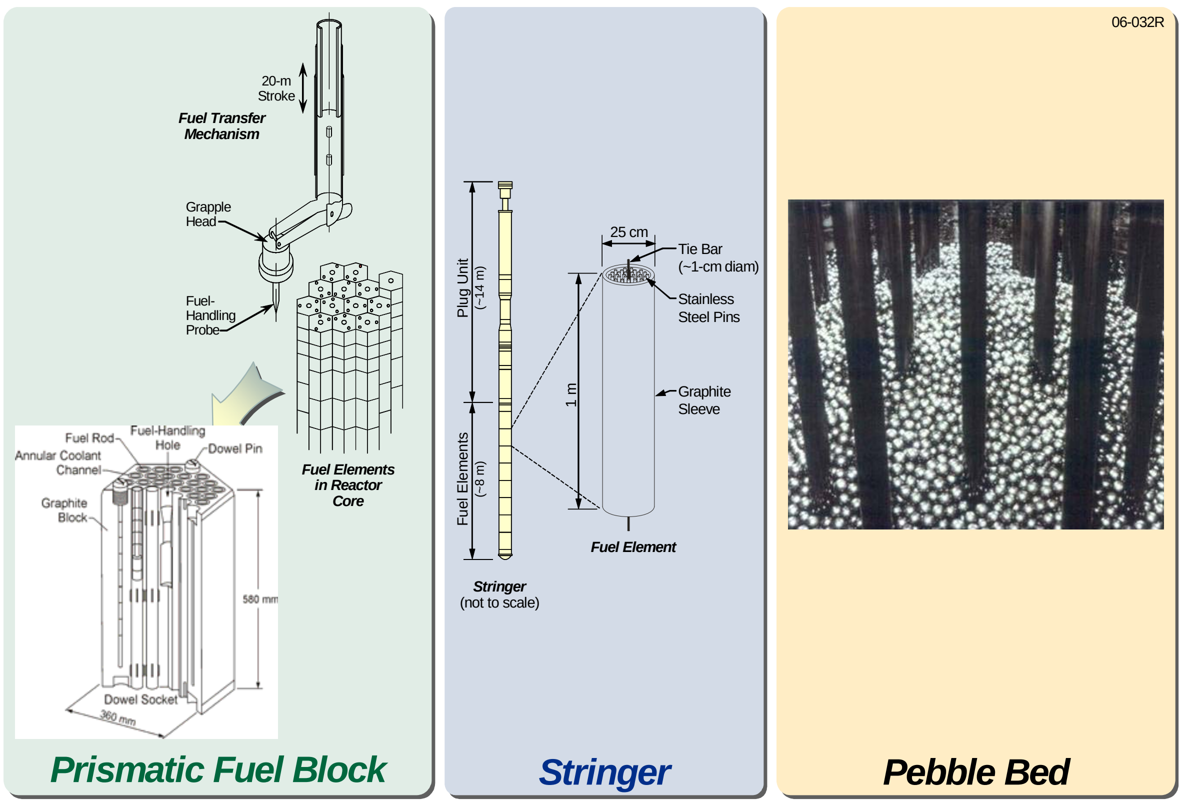

Fig. 4.1. High-temperature reactor fuel types and core configurations.

All of these fuel designs as currently envisioned for the LS-VHTR use coated-particle fuels in which the uranium oxide, carbide, or oxycarbide fuel is in the form of small microspheres that have multiple coatings of carbon and silicon carbide. These coatings act as a high-temperature equivalent of the metallic cladding found in more-traditional fuels. The microspheres are then incorporated into some type of graphite matrix, which can be in one of many geometric forms. This fuel is the only type that has been demonstrated to be capable of operations at high-temperatures with high burnups for extended periods of time. The fuel can operate at normal conditions without failure at temperatures up to ~1250ºC for extended periods of time and up to 1600ºC under accident conditions without failure for limited periods of time. It is the coated-particle fuel in the graphite matrix that makes high-temperature reactors viable. Graphite-based fuel and graphite components are chemically compatible with two coolants: (1) noble gases, such as helium, and (2) fluoride liquid salts.

4.1 PRISMATIC-GRAPHITE-FUEL HIGH-TEMPERATURE REACTORS

The graphite-matrix coated-particle fuel can be incorporated into prismatic graphite blocks (Fig. 4.2) that are the fuel assemblies for HTGRs. Several variants of this specific fuel form exist. The graphite block has coolant channels. Typically, the microspheres are incorporated into graphite fuel compacts in which the microspheres are mixed with graphite powder and compressed. The compacts are then placed in fuel holes drilled into the prismatic graphite block.

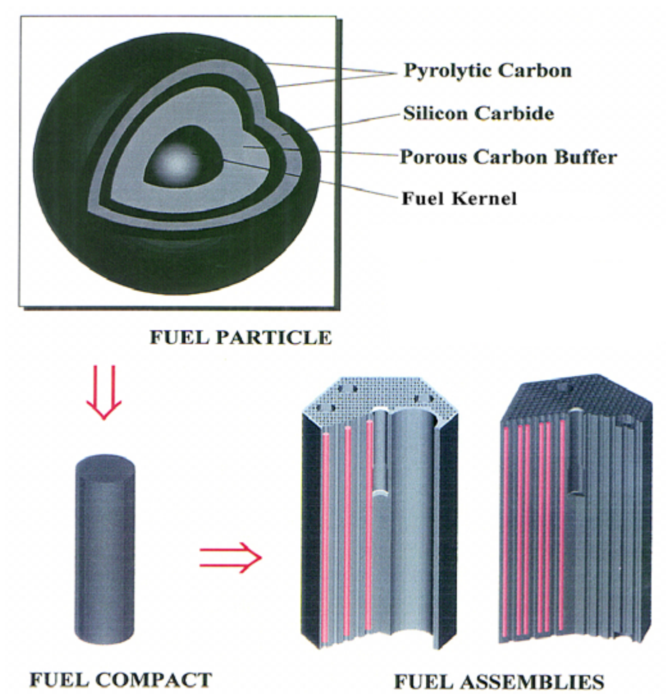

Fig. 4.2. Prismatic high-temperature fuel.

The fuel is in the form of prismatic assemblies to allow the blocks to lock together into a reactor core. Typical fuel-block dimensions are 36-cm width across the flats and 79-cm height. The reactor core is a three-dimensional stack of blocks. This geometric form allows three-dimensional refueling strategies that maximize fuel burnup and flatten the power distribution relative to the two-dimensional refueling strategies used in LWRs. The fuel form gives the reactor-core designer great freedom in choosing (1) the ratio of the fuel to the moderator and to the coolant and (2) the three dimensional neutronic characteristics of the reactor core. However, this freedom in core design options comes at the cost of more-complicated three-dimensional fuel-handling operations and a fuel assembly that also contains the moderator. To date, prismatic fuel has been the baseline for the LS-VHTR and thus the neutronics for prismatic fuels with liquid salt coolants has been studied more extensively than pebble or stringer fuel.

4.1.1 Prismatic-Fuel Gas-Cooled High-Temperature Reactors

Two helium-cooled high-temperature reactors have been built using prismatic fuel blocks: (1) the FSVR, which was built in Colorado and later decommissioned, and (2) the operating Japanese High Temperature Test Reactor (HTTR). The FSVR was a medium-size demonstration power reactor; thus, this refueling experience is most relevant to the LS-VHTR. In addition, eight large HTGRs were sold, with the orders later being cancelled. While these reactors were not built, the design and engineering studies associated with their refueling are directly applicable to the AHTR.

In terms of refueling, the FSVR (U.S. Nuclear Regulatory Commission, 1991) and the planned commercial reactors had similar design characteristics. The refueling was off-line at low pressure and relatively low temperatures. The technical specifications were <1 psig and helium-coolant reactor-core inlet temperatures of <74ºC (165ºF). The refueling machine (Fig. 4.3), which was mounted on top of the prestressed-concrete reactor vessel (PCRV), removed the SNF blocks via a robotic arm that was lowered from the refueling machine, withdrew the SNF from the reactor vessel, and then placed the SNF in a transfer cask. The operations were controlled by computer with operators monitoring the progress. This strategy results in short refueling outages for movement of the SNF. The SNF in the cask was then transferred to dry storage wells with water-cooled walls. To avoid the potential for fire if the SNF were exposed to air, the SNF temperature in the transfer cask and storage facilities was limited to ~400ºC (750ºF). Fresh fuel was then loaded in the reverse order.

Because refueling was performed on a PCRV with a limited number of penetrations, each refueling channel through the vessel head was used to refuel multiple columns of blocks in the reactor core. The large distance between the top of the reactor core and the top of the PCRV implies that HTGRs have large refueling machines. The FSVR refueling machine is shown in Fig. 4.4. While there were other mechanical difficulties with the FSVR, the refueling machine was successful and reliable. Studies (Paget, 1967) were initiated for development of online high-temperature refueling (see Sect. 4.3); however, with the cancellation of the high-temperature reactor program, online refueling for these gas-cooled reactors was not developed.

4.1.2 Prismatic-Fuel Liquid-Salt-Cooled High-Temperature Reactors

The LS-VHTR base-case prismatic-core design (Ingersoll et al., 2005) is very similar to that of the HTGR with a prismatic fuel. If it is assumed that the in-vessel fuel-handling machine can be constructed of the same advanced alloy as the reactor vessel, then it is reasonable to expect that the LS-VHTR design of the in-vessel fuel-handling machine could closely resemble the design proposed for the gas-turbine modular helium reactor (GT-MHR) machine [General Atomics, 1996], the follow-on gas-cooled reactor to the plants designed in the 1970s.

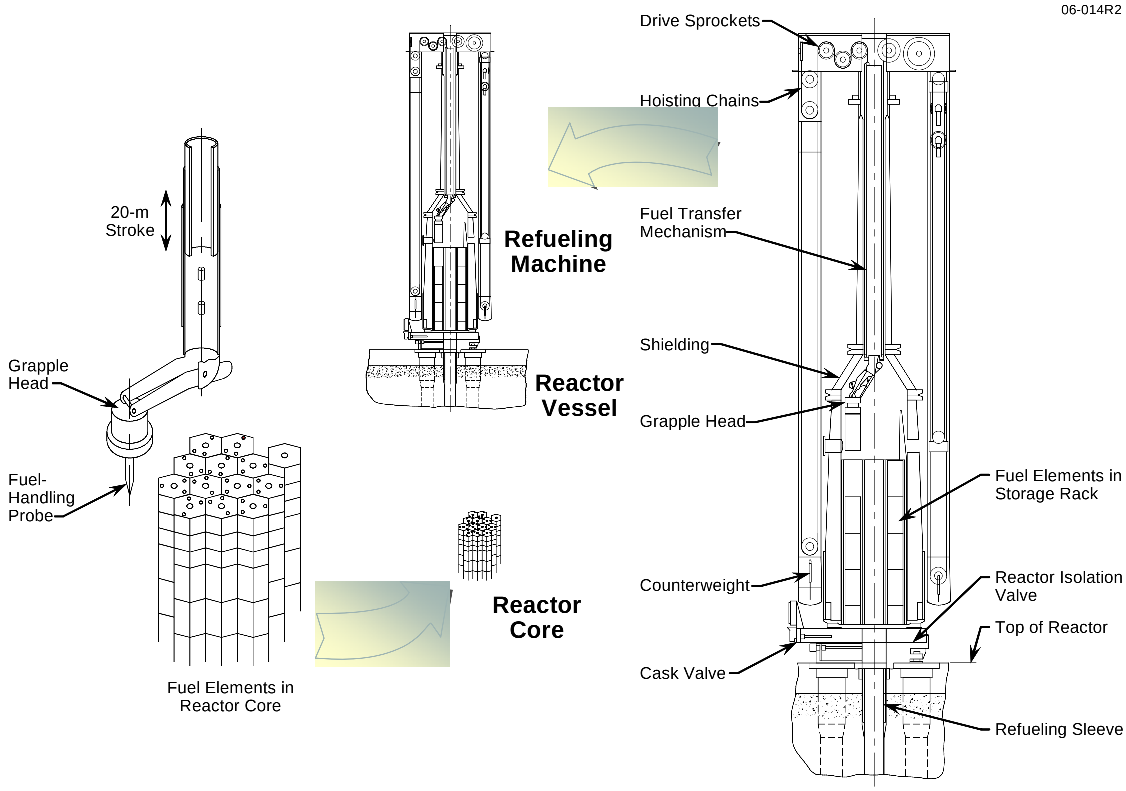

Fig. 4.3. Schematic of Fort St. Vrain refueling machine. Shown are in-vessel robotic arm over the reactor core; relationship of refueling machine, reactor vessel top, and reactor core; and refueling machine placing fuel element in transfer cask.

The number of fuel elements to be moved may be significantly less in an LS-VHTR compared with the historical experience with large prismatic-fueled gas-cooled reactors.

- *Plant efficiency*. The earlier HTGRs used a steam cycle that had a lower electricity-to-heat ratio than that of the LS-VHTR. This lower efficiency implies that these gas-cooled reactors require (1) more thermal energy, (2) more fuel assemblies, and (3) more refueling operations per unit of electrical. As shown in Table 4.1, the number of fuel blocks that must be handled per unit power output is expected to be significantly less for the LS-VHTR compared with those for the earlier gas-cooled reactors.

- *Block length*. The prismatic fuel block height of early gas-cooled reactors was based on the limitations of graphite manufacturing and machining in the 1970s. Since then, manufacturing methods have produced better graphites with more uniform properties. Similarly, major advances in manufacturing capabilities, such as the automated drilling of holes, have occurred. As a consequence, expert opinion indicates that the fuel block height could be increased by a factor of 2 to 3, affording the potential to significantly reduce the number of fuel elements per unit power output. Trade studies would be required to determine the optimum fuel element height.

Fig. 4.4. Photograph of Fort St. Vrain refueling machine.

Table 4.1. Comparison of FSV, commercial HTGR, and LS-VHTR reactor cores

Property	FSVR	Commercial HTGR[a]	LS-VHTR
Gross power [MW(t)]	842	3000	2400
Net power [MW(e)]	342	1160	1300[b]
Number of fuel columns	259	493	265
Number of fuel blocks per column	6	8	10
Total number of fuel blocks	1554	3944	2650
Number of blocks per MW(e)	4.5	3.4	2.0

[a]Commercial HTGRs were ordered and partly designed but not built.
[b]With a multi-reheat helium-cooled Brayton power cycle, the power output is 1357 MW(e) for salt exit temperatures of 1000ºC and 1235 MW(e) for salt exit temperatures of 800ºC. Several coolant exit temperatures are being considered.

Beyond the obvious difference that the salt-cooled reactor will refuel at higher temperatures in liquid salt, potentially significant differences are noted for refueling HTGRs when compared with LS-VHTRs.

- *Refueling machine size*. The LS-VHTR is a low-pressure reactor with a flat vessel lid above the reactor core; the distance between the top of the reactor core and the top of the reactor vessel is only a few meters. The height of the refueling machine (assuming the same approach is used) will be short in comparison with HTGRs with prismatic fuel. This factor will simplify some of the mechanical features.

- *Alternative fuel transfer options*. The LS-VHTR offers the unique refueling option of horizontal transfer of the SNF to a storage facility through a horizontal transport port near the top of the reactor vessel. This option is possible because of two characteristics of this design: (1) short SNF block height (small hole size in the reactor vessel) and a low-pressure reactor in which horizontal transfer ports through the reactor vessel wall are a potentially viable option. This option is described below.

- *Fuel hold-down*. The average fuel density may be less than or greater than that of the salt. If the fuel density is less than that of the salt (such would be the case with Fort St. Vrain type fuel), ballast or hold-down mechanisms are required to hold the fuel in place—as are used in a number of other types of reactors.

A variety of options are available to hold fuel in place. One or more of these features would likely be adopted if prismatic fuel is used.

- *Lock-down mechanisms*. One option is a mechanical locking device on the bottom of each fuel block that would engage the fuel-handling indent on the next lower fuel block. The bottom reflector block would be locked to the vessel bottom. The grapple portion of the device would closely resemble the grapple on the in-vessel fuel-handling machine. Such a device could be engaged and released by mechanical linkages manipulated by simple vertical motions of the in-vessel handling machine. The device could be constructed of high-strength high-temperature alloy (molybdenum, etc.) or a carbon−carbon composite to withstand the high core temperature. At the same-time, this device could be relatively lightweight, since only modest strength would be needed to overcome the small buoyancy forces. Such a low-mass device would not significantly alter the nuclear or structural performance of the existing LS-VHTR design.

- *Ballast fuel element*. The fuel assembly density can be increased by addition of weights within the fuel assemblies. Potential weight options include zirconium, molybdenum, and other metals or compounds encapsulated in molybdenum or carbon−composite jackets. Ideally, the ballast would be a single slug near the bottom of the fuel assembly to minimize neutron absorption. For these options, the neutronic penalty depends on the height of the prismatic block, the choice of ballast, and other design features. However, it should be noted that any ballast will degrade the neutronics and cost studies will be required to determine efficacy.

- *Top-lock mechanism*. Locking mechanisms to hold the core in place can be installed above the reactor core. Figure 4.5 shows one set of options, in which the fuel is held down by a series of beams that slide sideways to allow access to one column of fuel assemblies at a time. (LWRs use a top core plate and fuel assembly springs to hold the LWR fuel assemblies in place.)

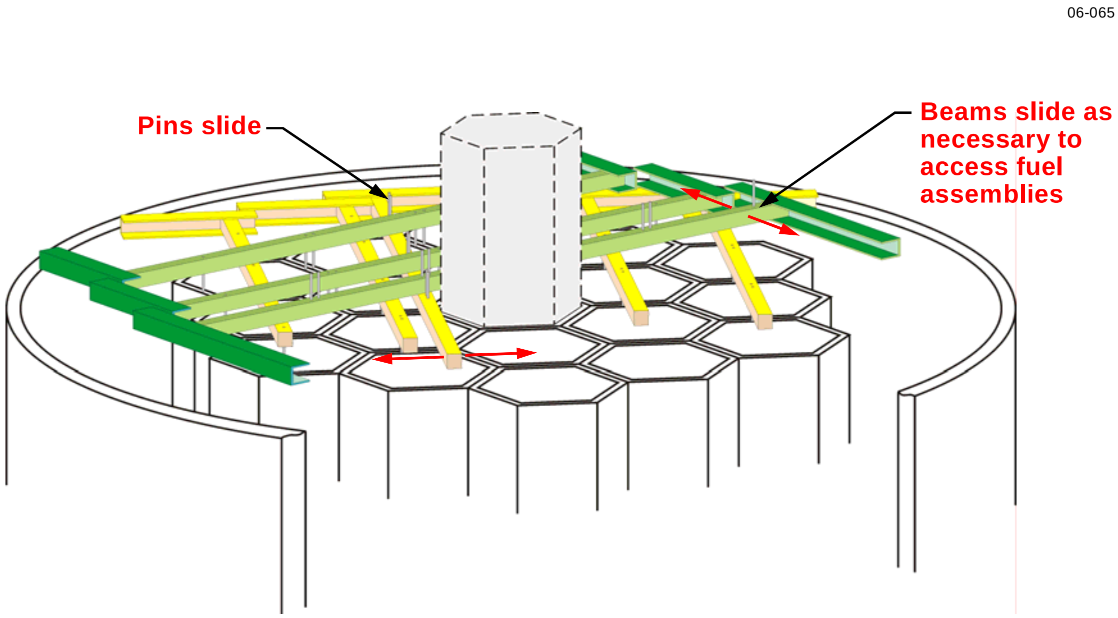

Fig. 4.5. Example of a top-locking mechanism for a prismatic-fuel reactor.

Several variants of the AHTR are being examined (Forsberg, 2006a). One variant (Peterson and Zhao, 2006) is the AHTR with metallic internals (AHTR-MI), which has a design strategy to minimize the number of high-temperature components. The proposed refueling method for this design variant is described as an example of an option for refueling a prismatic-fuel LS-VHTR. Figure 4.6 shows a vertical cross section of the reactor during normal operation and during refueling operations, while Fig. 4.7 shows a horizontal cross section. As in the FSVR, the prismatic fuel blocks (325 fuel columns, 8 blocks high) with control rods are offset downward 0.15 m to counteract shear motion across the core and maintain column alignment.

The initial steps in refueling are somewhat similar to those for the EBR-II (see Sect. 5). The control rod drives are disconnected, and the vessel lid is partly raised. Refueling of the AHTR-MI is performed using the same general approach that was developed and demonstrated for the FSVR, using a refueling machine with a grappling mechanism, and refueling the core by sector. However, unlike that for HTGRs, the AHTR-MI design includes a fuel transfer channel, shown in Fig. 4.6, which allows more-rapid fuel movement than is possible with the transfer cask systems used for HTGRs. The fuel elements do not require transfer to a shielded cask; they can be directly transferred to an intermediate storage facility that is located outside the primary reactor vessel.

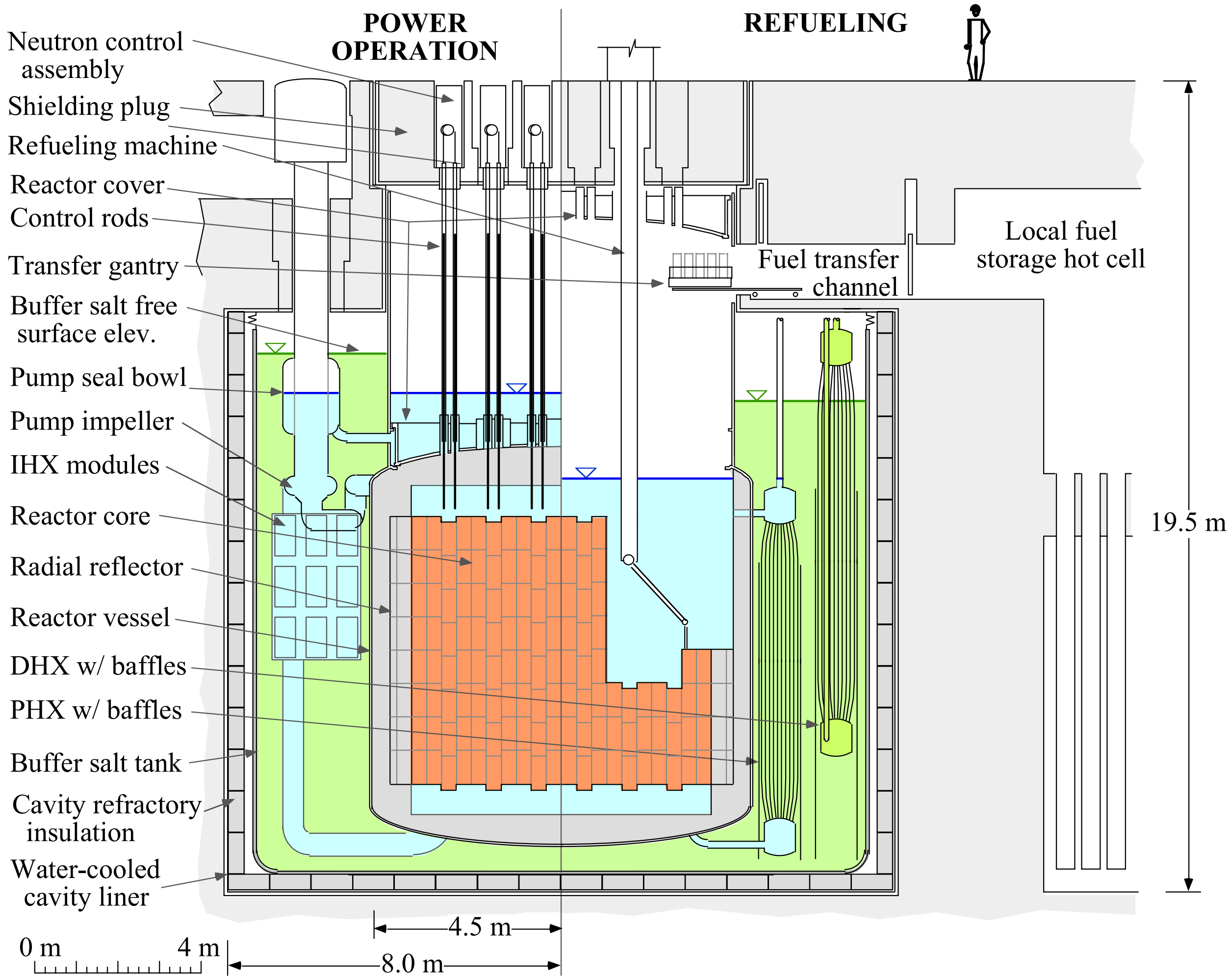

Fig. 4.6. Vertical cross-section of the AHTR-MI for normal operation (left) and refueling (right) modes. IHX = intermediate heat exchanger; DHX = decay heat exchanger; PHX = pool heat exchanger. (Source: Peterson and Zhao, 2006.)

This option has the potential to significantly reduce refueling time by avoiding SNF cask operations that include (1) transferring SNF to a cask, (2) closing the cask, (3) decoupling the cask from the reactor lid, (4) moving the cask to the SNF storage area, (5) coupling the cask to the SNF storage area, (6) opening the cask, (7) transferring the SNF to the storage area, and (8) each of these operations in reverse for the fresh fuel. This direct-transfer option exists because (1) the prismatic fuel blocks are short, which allows transfer of SNF via a relatively small horizontal penetration through the side of the reactor vessel, (2) the lower power density of the SNF reduces the need to cool the SNF during transfer operations, and (3) the reactor is a low-pressure machine that makes practical a horizontal transfer tube through the vessel wall.

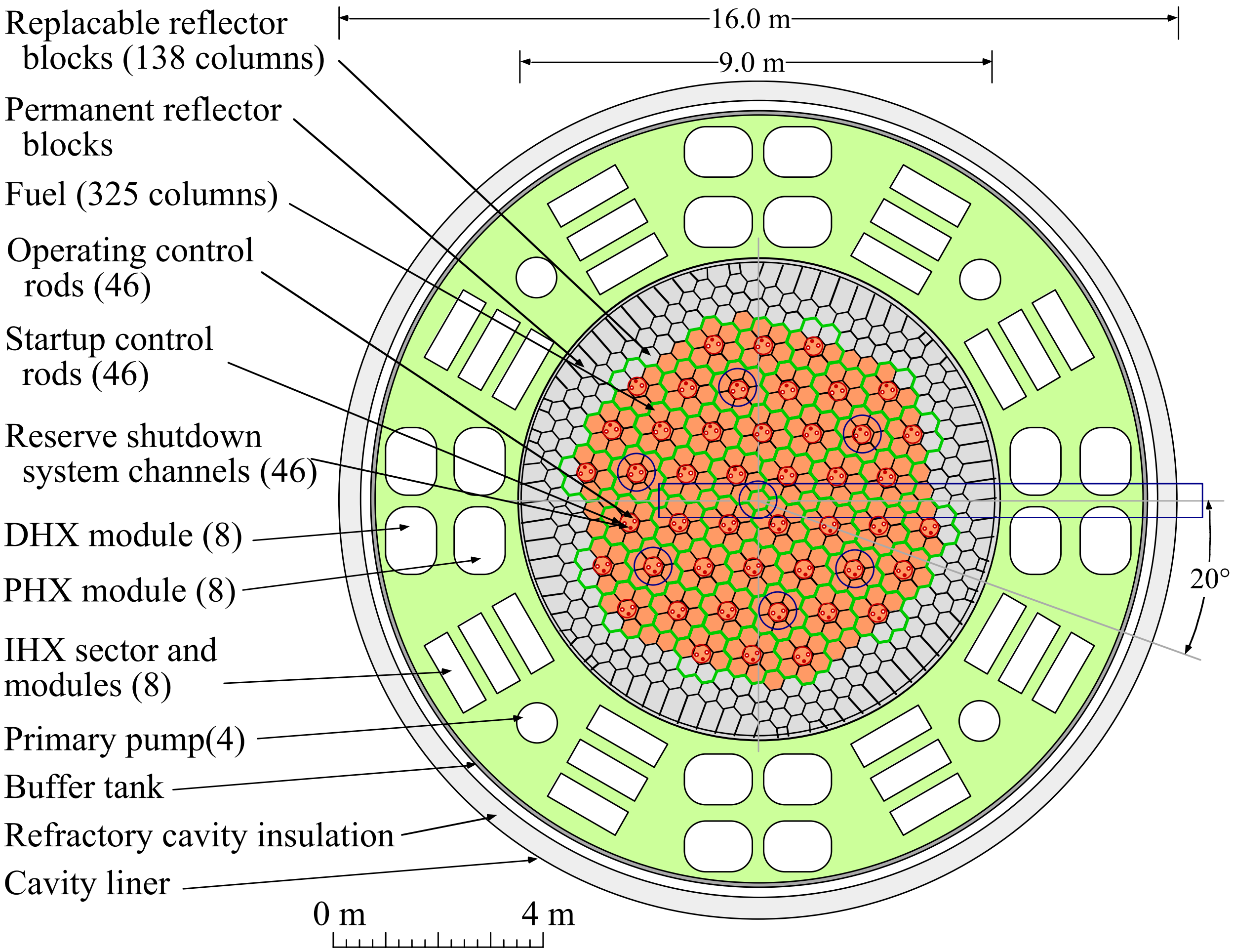

Fig. 4.7. Horizontal cross section of the AHTR-MI, showing the refueling port locations.
IHX = intermediate heat exchanger; DHX = decay heat exchanger; PHX = pool heat exchanger.

4.2 PEBBLE-BED REACTOR

PBRs use the coated-particle fuel in a graphite matrix compacted into pebbles—typically about 6 cm in diameter (Fig. 4.8). Current estimates indicate that pebbles have the lowest fabrication cost of any of the three fuel geometry options. The reactor core is a bed of pebbles. The THTR core is shown in Fig. 4.9. The vertical structures are channels for control rods.

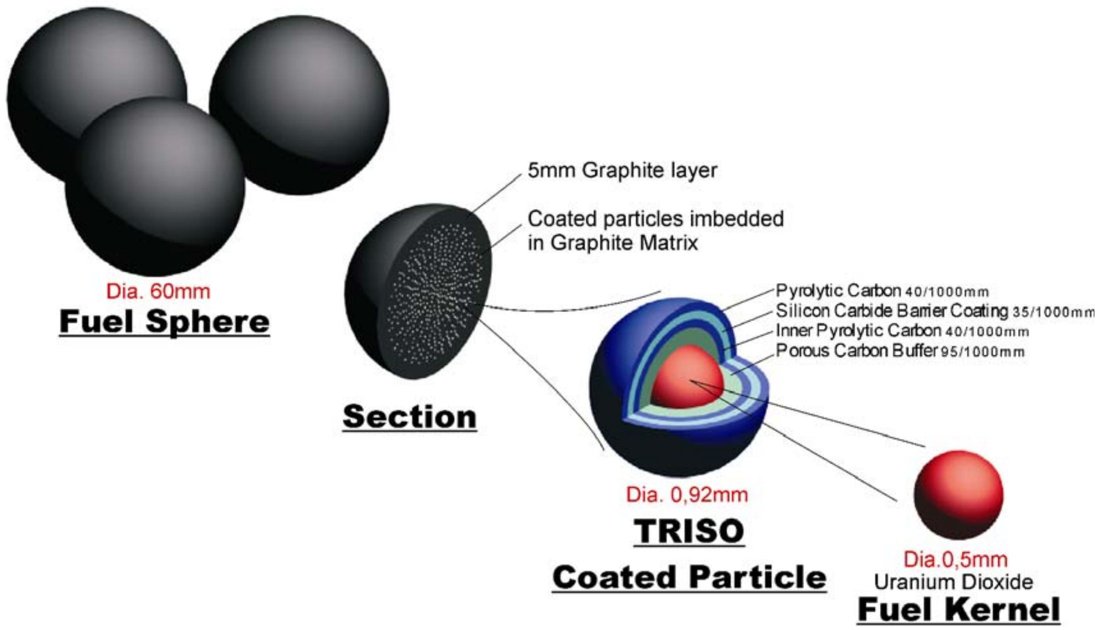

Fig. 4.8. Pebble-bed fuel.

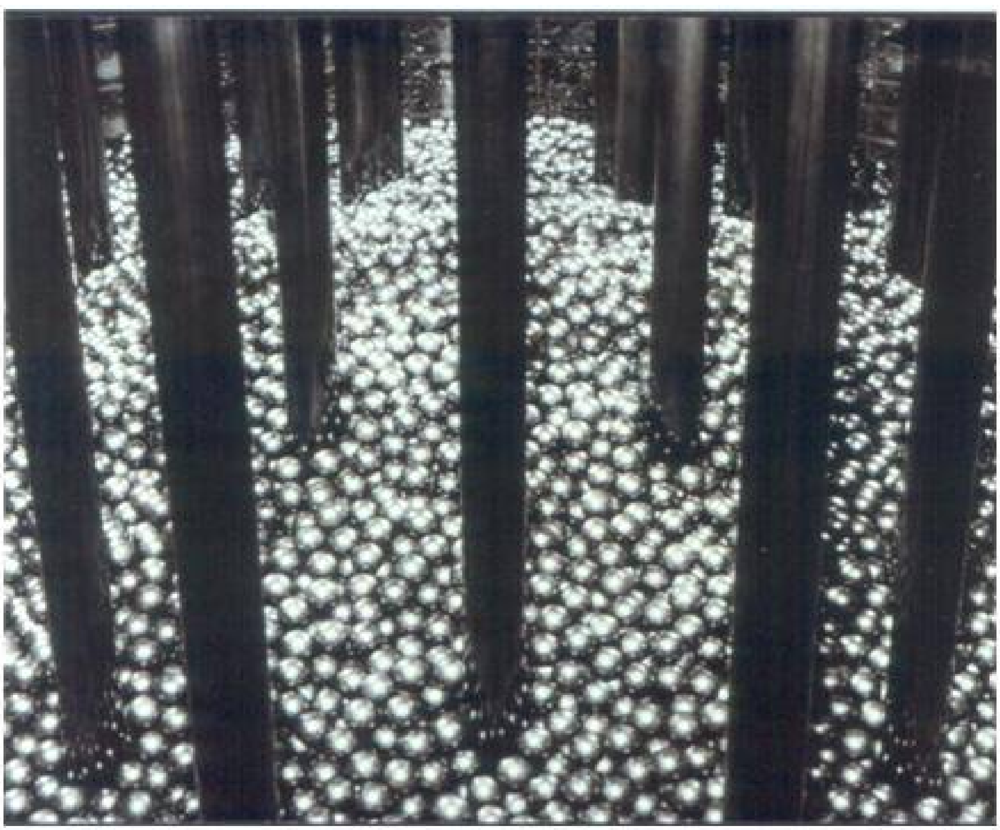

Fig. 4.9. German THTR core.

Two helium-cooled PBRs have been built, operated, and decommissioned in Germany, a small pebble-bed test reactor [the high-temperature reactor (HTR)] has been recently built in China, and a precommercial pebble-bed modular reactor (PBMR) is being built in South Africa (Table 4.2). These reactors are refueled online at temperatures up to 950ºC at full pressure. A slow continuous flow of pebbles occurs through the reactor core, with pebbles added at the top of the core and removed at the bottom. The pebbles go through the reactor core several times before being fully burnt. Extracted pebbles are sent through a radiation detector that determines burnup as well as the disposition of the pebble as SNF for disposal or for recycle back to the core for additional burnup. PBRs operate with very low excess reactivity and relatively low enrichments. A simplified schematic of the refueling system is shown in Fig. 4.10.

4.2.1 Pebble-Bed-Fuel Liquid-Salt-Cooled High-Temperature Reactors

Limited studies (deZwaan, 2005) have been performed on the large liquid-salt-cooled PBRs with power outputs from 2400 to 4000 MW(t). In addition, more detailed Russian studies have been conducted of a small [16 MW(t)] salt-cooled reactor for production of electric power [6 MW(e)] at remote sites. For most salt coolants, the pebbles will float under almost all conditions (at sufficiently high temperature, salt density may drop below the pebble density). This will alter the refueling operations (Fig. 4.11) compared with those for a helium-cooled PBRs. A salt-cooled PBR could have a zone under the reactor that contains vertical neutron absorbers. If the coolant level is decreased or the salt temperature increases sufficiently, the pebbles would go into a subcritical environment. A graphite outlet plenum structure with a metallic reactor cover would be located at the top of the reactor core to allow coolant flow out of the reactor core and collect the pebbles for transfer to the fueling machine. The pebbles flow toward the refueling machine rather than having the refueling machine reach into the reactor core to recover the pebbles.

With a salt-cooled PBR, the pebbles would be removed from near the top of the core and reinjected under the reactor core. Because the terminal rise velocity of the pebbles is quite low (<0.5 m/s), returned pebbles can be injected into the primary coolant piping to be carried to the reactor core, so the process of refueling may be greatly simplified. Furthermore, pebble transport hydrodynamics can be reproduced in scaled experiments using water and polypropylene spheres, where the Reynolds and Froude numbers, which govern pebble bed motion, are matched at approximately 50% geometric scaling. A schematic of one approach is shown in Fig. 4.12. The option also exists to add mass to the pebbles, causing them to sink in the coolant and permitting refueling via a process similar to that for the existing PBRs.

4.3 STRINGER OR ASSEMBLY HIGH-TEMPERATURE REACTORS

The AGRs are graphite-moderated carbon dioxide−cooled high-temperature reactors that use stainless-steel-clad oxide fuel assemblies. Because of advances in technology, a liquid-salt cooled variant of these reactors appears feasible, provided that the metal components of the fuel assembly are replaced with carbon composite materials. The AGRs are described, as well as a potentially suitable fuel assembly for a liquid-salt variant.

4.3.1 Stringer or Assembly Gas-Cooled High-Temperature Reactors

The United Kingdom has built and currently operates 14 AGRs. The 14 plants are similar but not identical. An example is the two-unit Dungeness B station. The first unit started operation in 1983 with the second unit coming online in 1985. Each reactor has an electric power output of 555 MW(e). The bulk gas exit temperature is ~640ºC, with a peak channel temperature of ~750°C. The operating pressure is ~43.3 bar. The reactors have steam power conversion cycles.

Table 4.2. Pebble-bed reactors[a]

Characteristic	AVR	THTR	HTR	PBMR
Country	Germany	Germany	China	South Africa
Initial Operation	1967	1984	2004	2011
Shutdown	1988	1990	Not applicable	Not Applicable
Heat [MW(t)]			10	400
Power [MW(e)]	15	300	Not applicable	165
T_{out} (°C)	950	750	700	900
T_{in} (°C)	270	250	250	500
Pressure (bar)	11	40	30	90

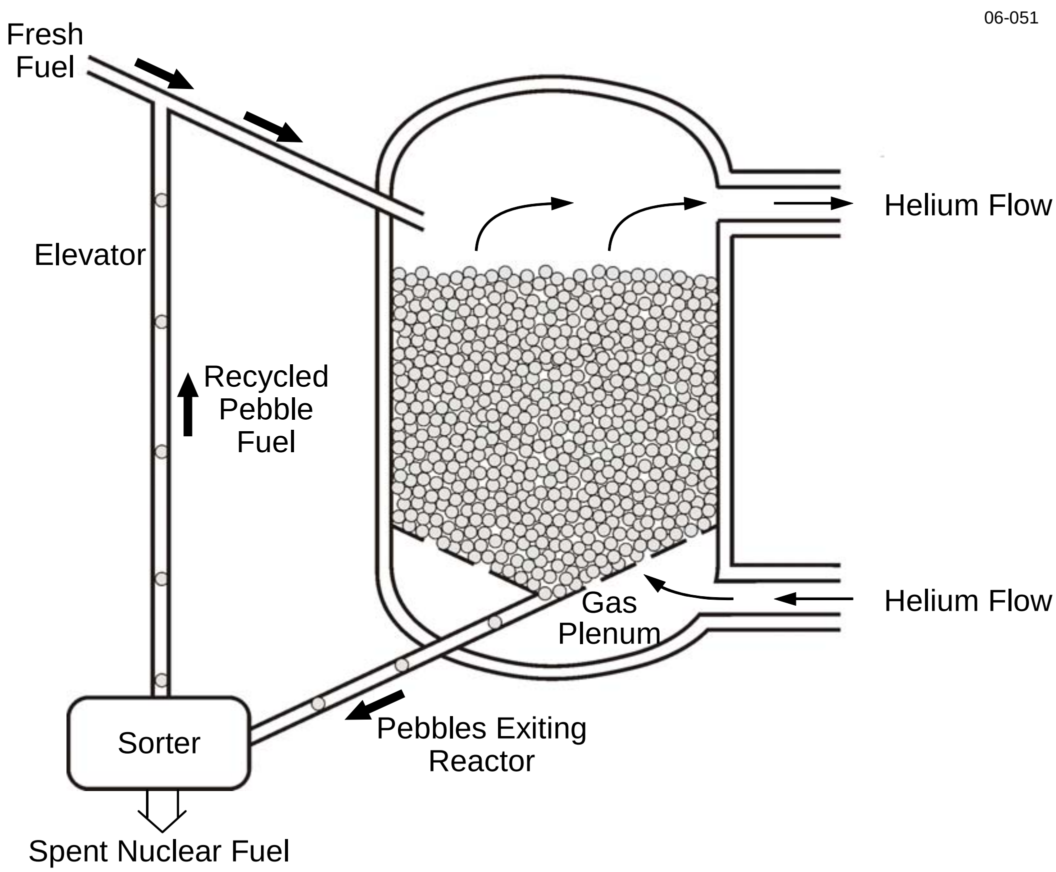

Fig. 4.10. Refueling of a helium-cooled pebble-bed reactor.

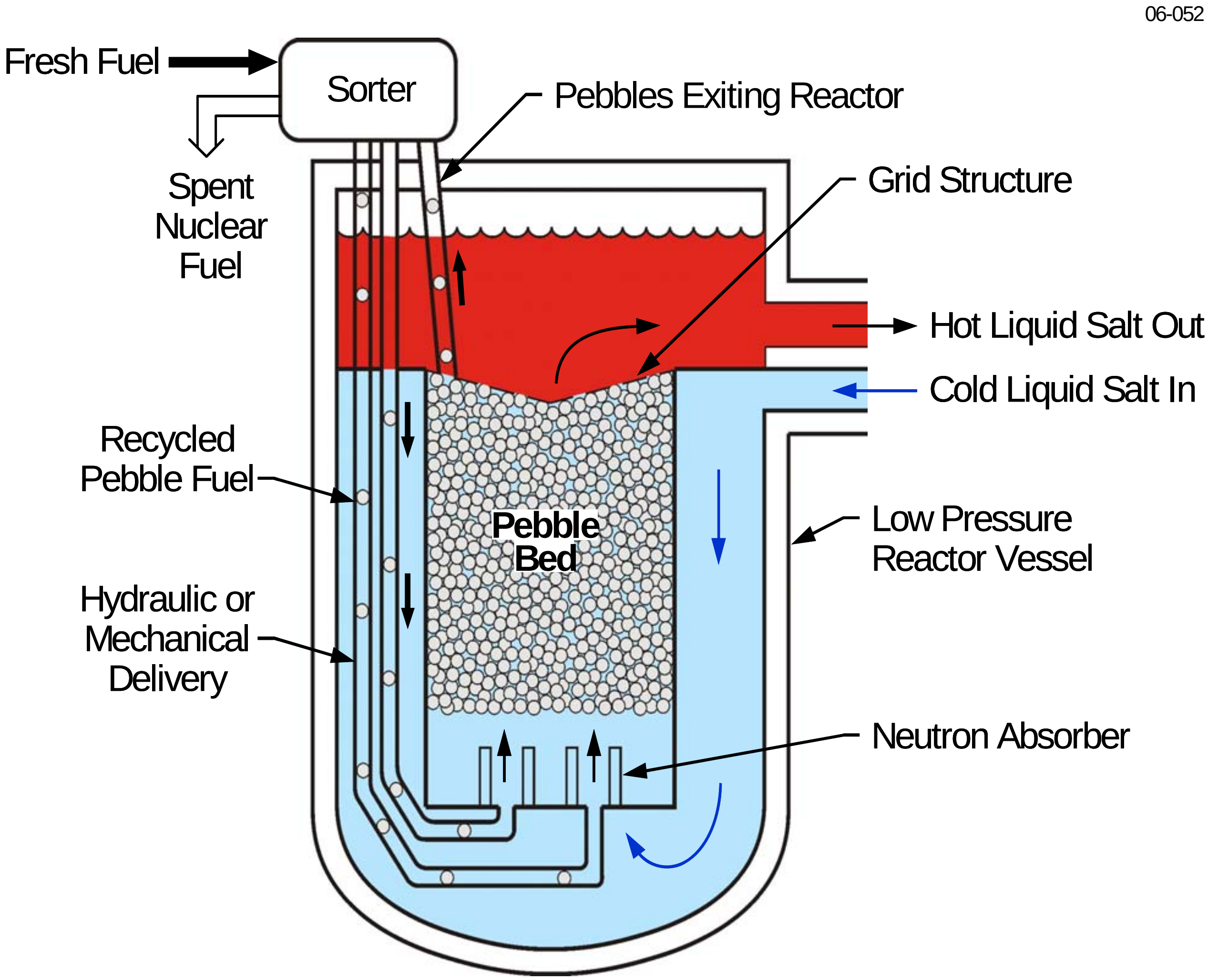

Fig. 4.11. Refueling of a salt-cooled pebble-bed reactor.

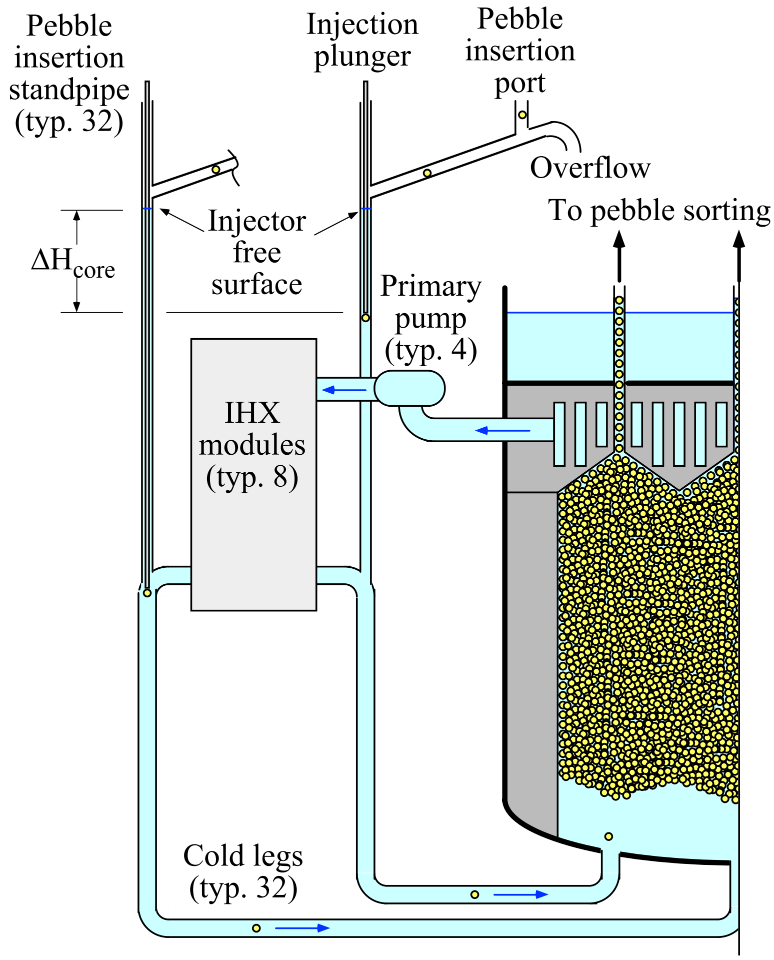

Fig. 4.12. Schematic of pebble recirculation system being studied at the University of California at Berkeley, showing pebble insertion into the coolant cold legs for injection at 32 locations around the bottom inlet plenum of the reactor.

The graphite AGR core is located within a thin steel structure with thermal insulation that, in turn, is located in a large PCRV. The reactor vessel has vertical holes for fuel stringers (Fig. 4.13). A stringer consists of multiple fuel assemblies, neutron moderator sections, radiation shielding, thermal insulation, pressure seals, and other components. It extends from the bottom of the reactor core to the top of the PCRV. The stringer includes, at the bottom, eight 1-m-long fuel assemblies with a graphite sleeve and 36 stainless steel−clad fuel pins with uranium dioxide fuel pellets. A 1-cm nimonic® alloy PE16 tie bar goes through each fuel assembly and holds them together as a single unit on a stringer. The graphite sleeve provides a gas flow channel, serves as part of the assembly with the grid structure that holds the fuel pins in the proper geometry, and provides some radiation shielding to reduce the rate of radiation damage to the permanent graphite in the reactor core. The sleeve is part of the SNF and is separated from the SNF pins for the purposes of disposal. It is not reused.

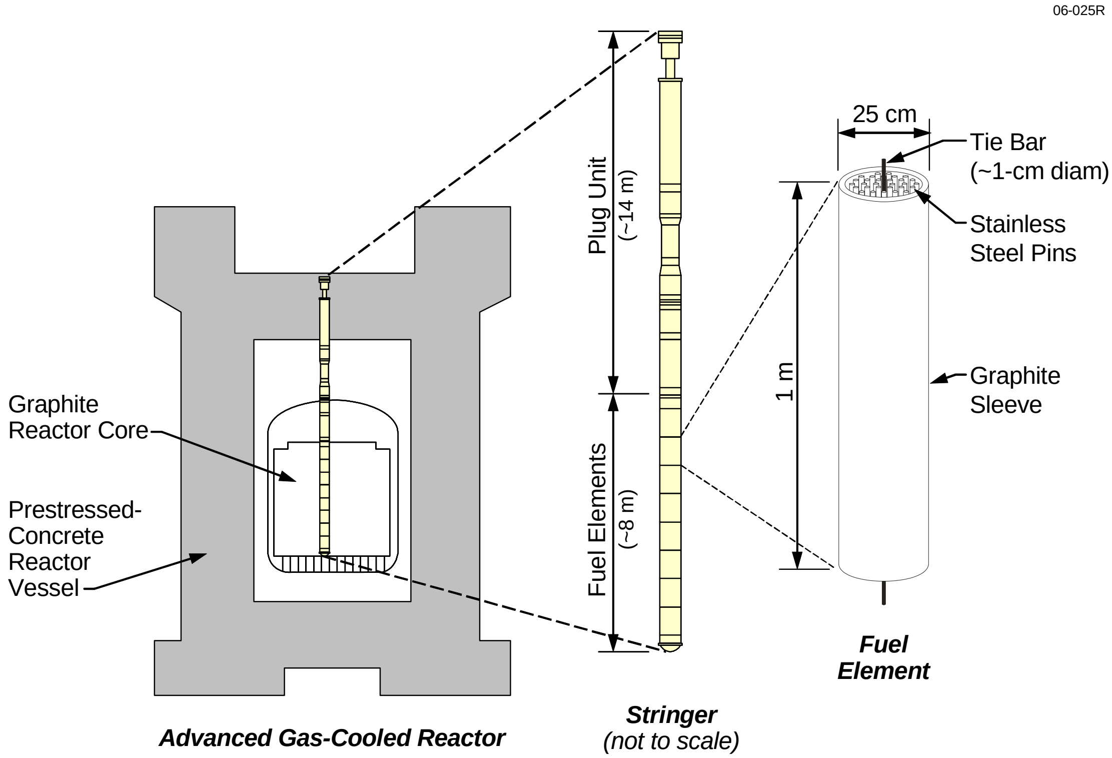

Fig. 4.13. Advanced Gas-Cooled Reactor, stringer, and fuel element.

The reactors were designed to be refueled online with a refueling machine located on top of the PCRV (Mottershead et al., 1995; Cornell et al., 1995; Dixon and Penny, 1995). The refueling machine couples to the reactor vessel with seals between the high-pressure PCRV and the high-pressure components in the refueling machine. SNF stringers, which extend from the bottom of the reactor core to the PCRV pressure boundary, are removed as a single piece from individual channels in the reactor core. The Dungeness B refueling machine is shown in Fig. 4.14. There is one machine for the two reactors.

Because the reactor is at power during refueling, the SNF assemblies must be cooled when they are being removed from the reactor. After an SNF stringer is removed, a new fuel stringer is inserted into the reactor. The use of stringers minimizes the number of components of the refueling machine subject to high temperatures and minimizes in-vessel operations. After removal of a stringer, the refueling machine decouples from the reactor and the fuel stringer is then transferred to the fuel storage area. After 30 days of cooling, the stringer is disassembled via separation of the individual fuel assemblies, which are then sent to storage. During online refueling operations, the machine refuels at high pressure (43.3 bar), with average gas outlet temperatures between 620 and 640ºC and peak temperatures approaching 750ºC in the hot channels.

Fig. 4.14. Dungeness B Advanced Gas-Cooled Reactor. (Courtesy of British Energy.)

When the plants were originally built, the AGRs were refueled online at full power for several years. A cracked graphite sleeve in one of the Hinkley Point B reactors led to a change in the refueling strategy. At a certain point when fresh fuel was being loaded into the operating reactor, the gas pressure inside the graphite sleeves in the reactor core exceeded the external coolant pressure, leading to tensile stresses and cracking in the graphite sleeve. The evaluations indicated that the full-power online refueling was practical but that the cost of in-core retrofits to modify the gas flow dictated different solutions for different stations. The 14 AGRs are not identical. Some of the AGRs are now refueled off-line, while others are refueled at full temperature and pressure but with the reactors at partial load—typically 25 to 30% of full power.

4.3.2 Stringer- or Assembly-Fuel Liquid-Salt-Cooled High-Temperature Reactors

The core design and stringer approach is potentially applicable to the LS-VHTR, as shown in Fig. 4.15, with the fuel assembly shown in Fig. 4.16. The AGR fuel assembly would be replaced by an all-carbon fuel assembly capable of very high temperature operation. In the prismatic-fuel Japanese HTTR (Fig. 4.17), the fuel microspheres are contained in graphite compacts; the compacts, in turn, are placed in graphite tubes. The tubes are mounted inside individual coolant channels of the prismatic fuel block. This arrangement maximizes heat transfer from fuel to coolant and is used to minimize peak fuel temperature. A variant of these fuel pins could be used to replace the fuel pins in an AGR fuel assembly. The stainless steel grid structure that holds the AGR fuel pins in place and the tie rod could be replaced by carbon−carbon composites. A preliminary assessment of the carbon−carbon technology and the AGR design has not identified any insurmountable fabrication challenges to create an equivalent carbon−carbon composite of the AGR fuel assembly. However, only limited analysis of such fuel designs has been performed. A significant fuel development effort would be required.

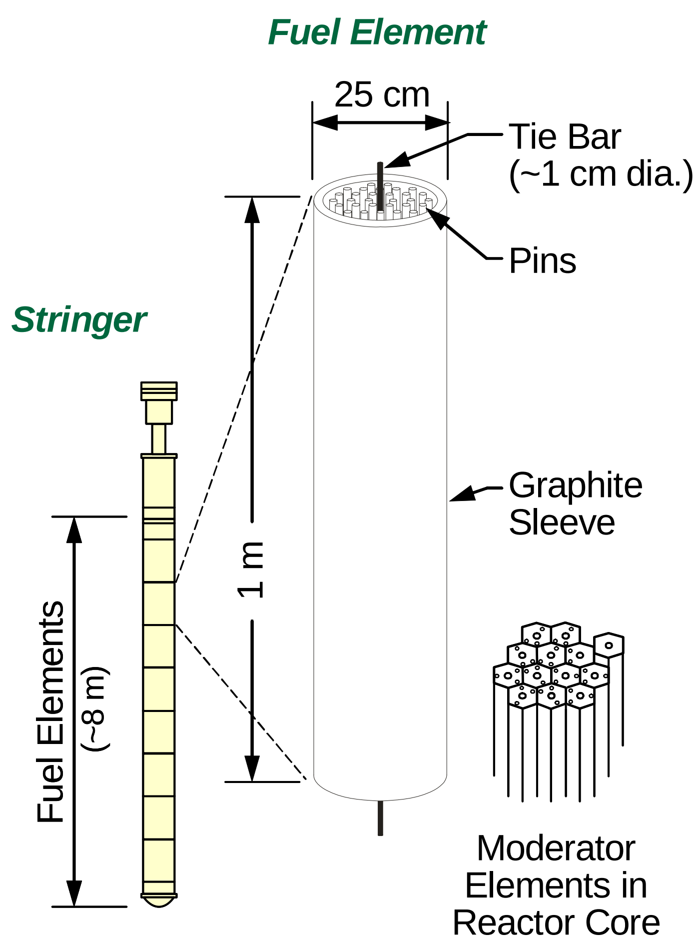

Fig. 4.15. Stringer for an LS-VHTR.

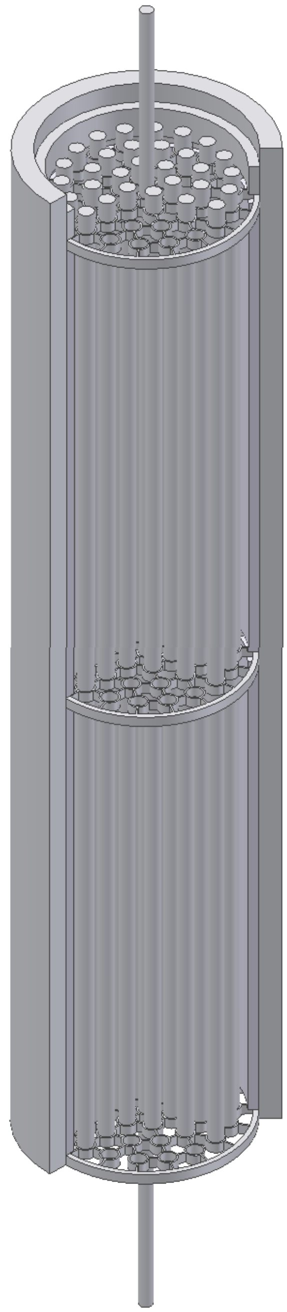

Fig. 4.16. Stringer fuel assembly for an LS-VHTR.

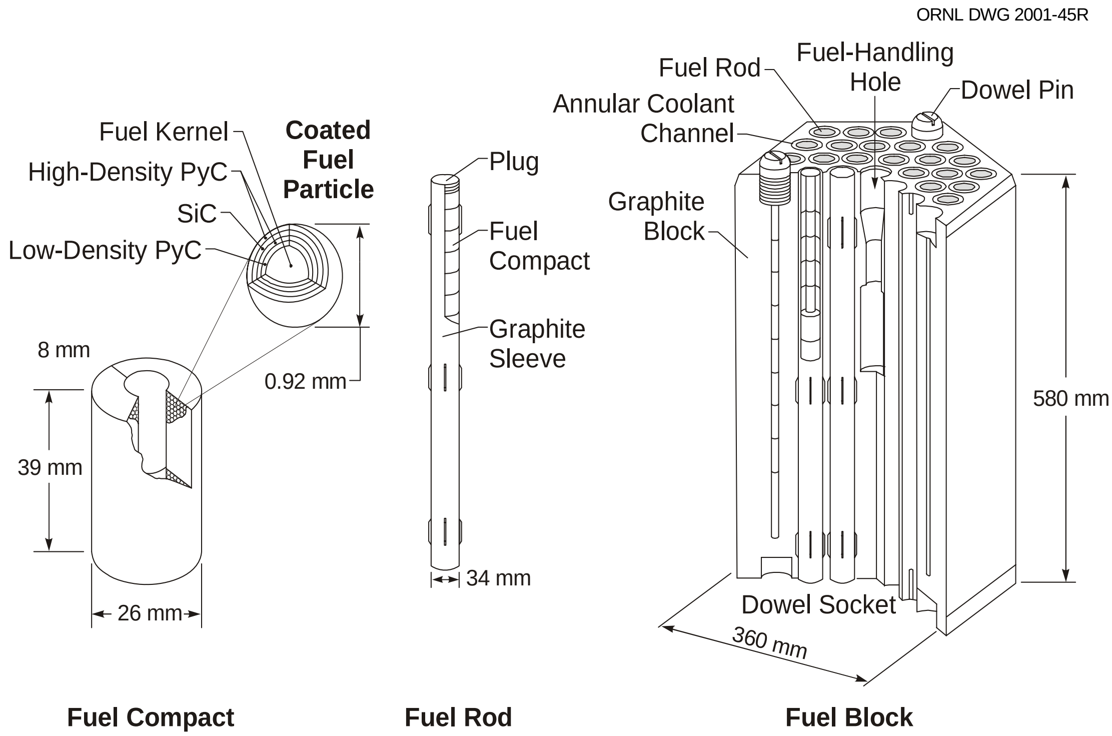

Fig. 4.17. Japanese High-Temperature Test Reactor fuel assembly with fuel rod.

The LS-VHTR variant of the AGR would retain the prismatic graphite moderator blocks; however, the graphite blocks would contain no fuel. Instead, each graphite block would have a central hole for a fuel stringer (Fig. 4.18). The graphite moderator blocks receive some radiation damage and would likely have to be replaced once or twice over the lifetime of the reactor; however, the graphite replacement would be relatively infrequent compared with the number of SNF refueling operations. Furthermore, the reactor could be defueled for graphite replacement with the salt level lowered during these operations. Such operations would simplify graphite replacement.

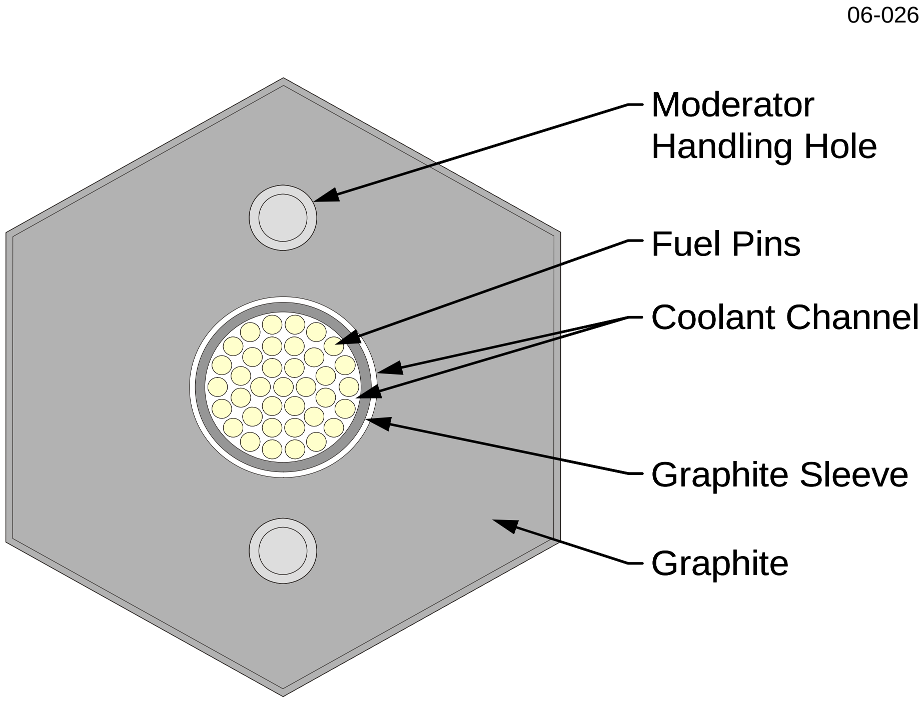

Fig. 4.18. Moderator block with stringer fuel assembly.

5. FAST REACTOR EXPERIENCE

5.1 COMMON SODIUM AND SALT REFUELING CHARACTERISTICS

Excluding military and space reactors, approximately 20 sodium-cooled fast reactors have been built in a variety of sizes and configurations. These vary from small test reactors to the French Super-Phenix plant, which had an output of 1240 MW(e). In the United States, several fast reactors were built. These included the EBR-II and the Fast-Flux Test Facility (FFTF)—a 400-MW(t) reactor. The Clinch River Breeder Reactor Plant (CRBRP), a commercial demonstration reactor, was designed and partly built before being cancelled. These machines provide a large experience base in refueling operations (Romrell et al., 1989; Althaus and Brahy, 1987).

The facility designs and many of the refueling characteristics of the LS-VHTR are similar to those of sodium-cooled fast reactors. As a consequence, much of the refueling technology for sodium-cooled fast reactors is applicable to an LS-VHTR with relatively minor changes required. This technology is reviewed herein. The reactors have the following similarities.

- *Low pressure*. Both types of reactors use coolants with vapor pressures significantly <1 atm. The low pressure allows the use of reactor vessels with flat lids and rotating plugs in the lids. These rotating plugs allow refueling machines to be positioned above any location in the reactor core—unlike high-pressure gas-cooled reactors, in which the access is limited. This is a major advantage for refueling and maintenance operations.

- *Controlled atmosphere*. Sodium reacts rapidly with air and water, while liquid salts react very slowly (see Sect. 3). In both cases, an inert atmosphere must be maintained above the liquid coolant to ensure coolant purity. Argon has been traditionally used as the cover gas in fast reactors because it is chemically inert, inexpensive, and dense relative to air. A heavy gas will remain in place whereas a light gas such as helium tends to move upward through the seals in the reactor-lid plugs while air tends to move downward. It is likely that argon will also be the cover gas for the LS-VHTR for similar reasons. Both sodium and liquid-salt vapors can "freeze out" on cold surfaces. This is avoided by the use of inert-gas purge flows to keep joints (such as in the rotating plug) free of frozen coolant.

- *High temperatures*. Both reactors operate at high temperatures. The peak refueling temperatures in some sodium-cooled fast reactors are similar to those of the LS-VHTR. The EBR-II in Idaho had a nominal refueling temperature of 370ºC, whereas the FFTF at the Hanford site in Washington had a nominal refueling temperature of 315ºC. However, the peak allowable fuel-handling temperatures for moving SNF from in-vessel storage to out of the vessel (Cabell, 1980; FFTF, 1983) were much higher. For the FFTF, the allowable peak element fuel temperature during handling operations was 538°C (1000°F). The high temperature has several implications for both sodium-cooled and salt-cooled reactors. For example, all motors and control equipment associated with refueling and maintenance are located outside the reactor vessel in a low-temperature environment with mechanical linkages to refueling equipment inside the reactor vessel. The fresh fuel must be preheated before insertion into the reactor vessel. Insulation becomes a major design requirement.

5.2 FAST REACTOR REFUELING EXPERIENCE

The design and performance characteristics of the EBR-II, FFTF, and CRBRP fuel-handling systems have been reviewed for potential relevance to the LS-VHTR (Cahalan and Taiwo, 2006). All these systems used liquid sodium as coolant. Fuel-handling operations were performed in a closed primary system at temperatures sufficiently high to maintain coolant liquidity and with an inert cover gas to avoid chemical reactions between the sodium and the air that contains oxygen and water. This summary provides brief descriptions of the fuel-handling systems and their operations, as well as discussion of design requirements relevant to the LS-VHTR.

The fuel-handling system designs for the EBR-II, FFTF, and CRBRP reflected the requirements imposed by the individual facility missions and reactor system designs. The EBR-II was intended as a developmental prototype for a breeder reactor power station. The FFTF was designed as an irradiation test facility for fuels and materials; and the CRBRP was to be a medium-sized demonstration commercial breeder reactor power station.

The pool-type EBR-II design features a large primary system tank containing the reactor and all primary coolant systems, including the reactor, the pumps, and the intermediate heat exchanger (IHX). This layout is similar to that for the LS-VHTR, which was described earlier. The EBR-II primary tank also contained an SNF storage basket that can hold a discharge SNF batch for an extended period (typically 100 days) of decay heat cooldown. The provision for in-vessel SNF storage simplifies ex-vessel handling and storage requirements.

The FFTF testing mission imposed requirements for relatively frequent refueling and handling of large test components, as well as handling of standard reactor fuel assemblies. The FFTF was designed as a loop-type primary system, with the reactor in a vessel connected by piping to the pumps and intermediate heat exchangers. The configuration limits in-vessel SNF storage capability and places additional requirements on ex-vessel fuel-handling equipment, relying on discharge to ex-vessel storage for routine cooldown operations.

The CRBRP was designed, licensed, and partly constructed before being cancelled. It was designed with strong reliance on FFTF experience, taking into account the CRBRP mission of power production, in contrast to the FFTF mission of fuel testing. The CRBRP mission required relatively infrequent refueling outages of minimum duration, during which a large fraction of the core would be refueled. The loop-type CRBRP primary system design provides only minimal in-vessel fuel storage; routine operations require immediate SNF discharge and transfer to ex-vessel storage for cooldown.

Temperature requirements for liquid-metal reactor fuel-handling system designs are typically set on the basis of maintaining spent fuel integrity (or test sample integrity for the case of the FFTF), and operational temperature requirements vary depending on mission. Actual system capabilities are set by materials compatibility and strength characteristics. In liquid-sodium systems, the structural material is typically stainless steel (316 or 304), which has excellent strength properties to 550^{o}C and beyond.

5.2.1 EBR-II Fuel-Handling System

The EBR-II fuel-handling system is designed to facilitate loading of fresh fuel into the reactor and removal of SNF from the reactor to the adjacent fuel cycle facility (EBR-II, 1971; Koch). The fuel-handling system is displayed in Fig. 5.1, which shows the reactor, the fuel gripper, and the hold-down mechanisms, the transfer arm, the storage basket, and the fuel-unloading machine.

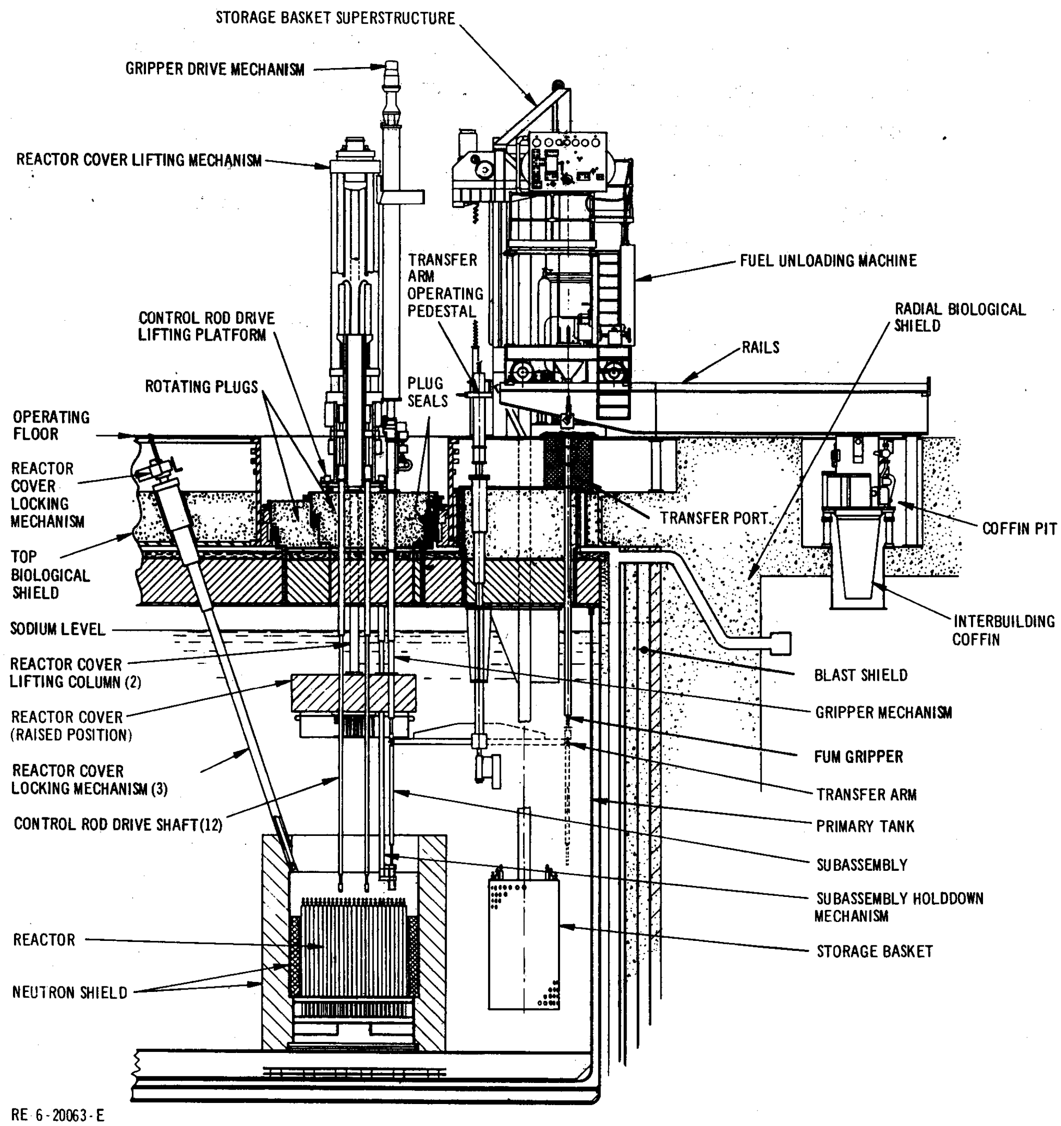

Fig. 5.1. EBR-II fuel-handling system.

During normal operation, the EBR-II reactor was enclosed within an inner vessel that caused hot coolant to flow from the reactor outlet plenum through a pipe to the IHX, where heat was transferred to the secondary coolant. Cold primary coolant exiting the IHX flowed into the primary tank, which held all the primary coolant, the reactor, the IHX, and two primary coolant pumps, as well as provide short-term SNF storage.

For refueling operations, the control rod drive shafts were disconnected and lifted, and the inner reactor vessel cover was raised. To remove a subassembly, the gripper was lowered to the reactor, where it grasped the subassembly to be lifted. Concurrently, the hold-down mechanism was lowered to prevent inadvertent extraction or movement of subassemblies adjacent to the gripped subassembly. The gripper drive mechanism raised the subassembly to be discharged above the reactor core, and plugs in the primary tank lid were rotated into a position in which the transfer arm mechanism could grasp the subassembly. During plug rotations, the hold-down mechanism provided support to the subassembly bottom, in addition to the top support provided by the gripper mechanism.

After grasping the assembly, the manually operated transfer arm rotated to locate the subassembly above one of three rows of positions in the storage basket, which rotated to provide correct positioning. For insertion of a subassembly into the storage basket, the transfer arm mechanism first lowered the subassembly partly into the basket and the storage basket was then raised to seat the subassembly. The transfer arm released the subassembly, and the storage basket was then lowered to the storage position.

Transfers of SNF to the fuel cycle facility from the in-vessel storage basket were conducted during reactor power operation. The SNF was stored for some time in the reactor vessel to reduce the decay heat loads. The transfer arm was used to remove a subassembly from the storage basket and move it to a location where it was grasped by the gripper of the fuel unloading machine. The fuel-unloading machine then raised the subassembly (1) to a dripping elevation, (2) to an argon-blowing station for additional sodium removal, and (3) into a shielded argon-cooled coffin through a transfer port in the primary tank cover. The rail-mounted fuel-unloading machine then traveled to the interbuilding transfer station, where it lowered the subassembly into a second interbuilding coffin, also cooled by argon. This coffin subsequently traveled to the fuel cycle facility. During all fuel-handling operations, the maximum fuel pin temperature was maintained below 650°C (1200ºF).

Loading of fresh fuel is accomplished with the inverse operation. Fuel subassemblies loaded into the reactor are first heated to 450°F (232°C) in the fuel-unloading machine prior to insertion into the primary tank sodium, which is maintained at 700°F (371°C).

5.2.2 FFTF Fuel-Handling System

The FFTF refueling system (Cabell, 1980; FFTF, 1983) includes facilities for the receipt, conditioning, storage, installation in and removal from the core of all core components (driver fuel assemblies, control assemblies) and test assemblies that are routinely removable. The reactor refueling system handled three types of core assemblies: 12-ft assemblies such as driver fuel; 40-ft assemblies such as fuels open test assemblies; and 40-ft assemblies such as materials open test assemblies.

An overall plan view of the FFTF reactor refueling facilities is shown in Fig. 5.2. The principal ex-reactor component is the closed loop ex-vessel machine (CLEM), shown in Fig. 5.3. The CLEM loads all components into the reactor vessel and removes all components from the reactor core. Fresh driver fuel and all SNF is transferred to and from the reactor in a core component pot (CCP) that can be inserted or removed through one of three fuel transfer ports in the reactor vessel top cover. The in-reactor components consist of three in-vessel handling machines (IVHMs) plus the three in-vessel storage modules. The FFTF requires three IVHMs because of closed test loops in the reactor core which interfere with direct access to the entire reactor core with one machine.

In the handling of 12-ft. assemblies such as driver fuel (Figs. 5.4 and 5.5), a fresh incoming assembly is first lowered by the crane into one of the two core component conditioning stations. In this station, the assembly is flooded with argon and heated to ~450°F (232ºC). The assembly is then picked up by the bottom-loading transfer cask and is transported, in an argon atmosphere, to the interim decay storage vessel. The assembly is then lowered into a sodium-filled CCP in the interim decay storage vessel. The assembly is held in liquid sodium at 500°F (260°C) to 600°F (316°C) until the reactor is shut down for refueling.

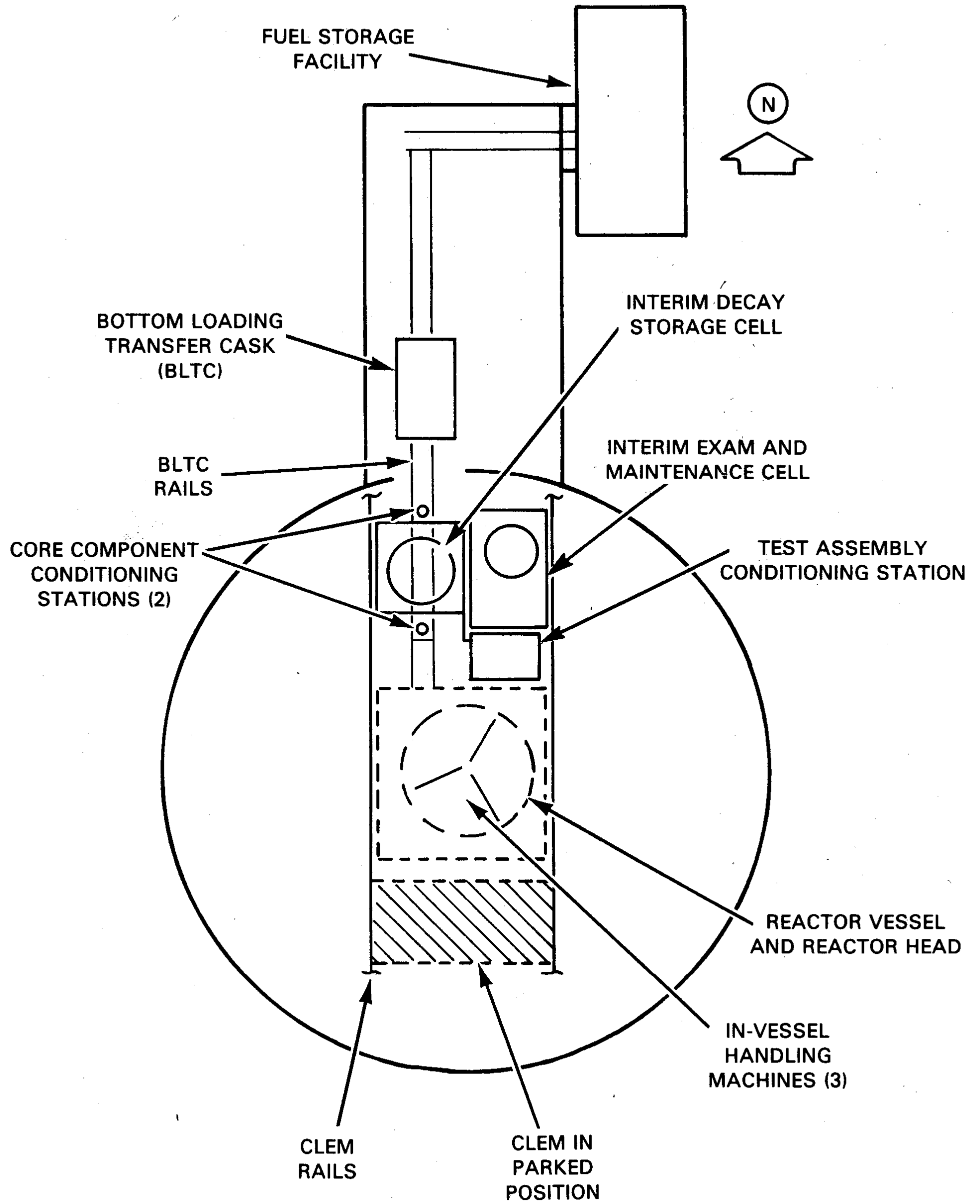

Fig. 5.2. General arrangement of FFTF reactor refueling facilities.

Fig. 5.3. FFTF closed loop ex-vessel machine (red machine on rail tracks).

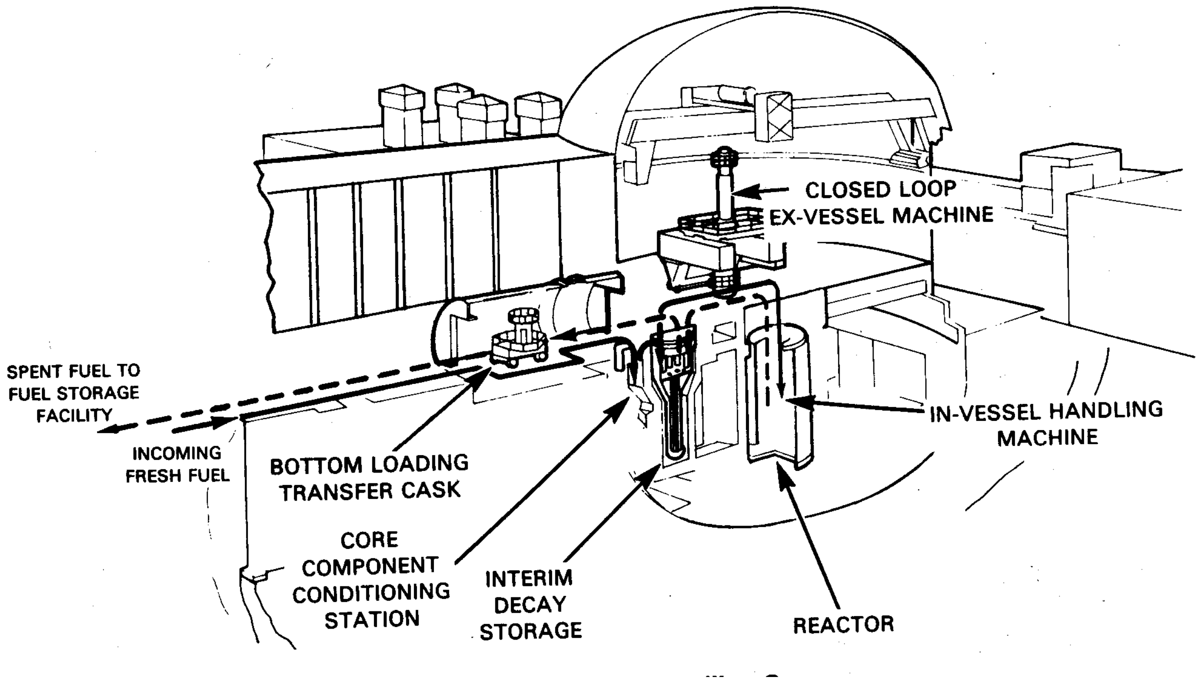

Fig. 5.4. FFTF driver fuel-handling sequence.

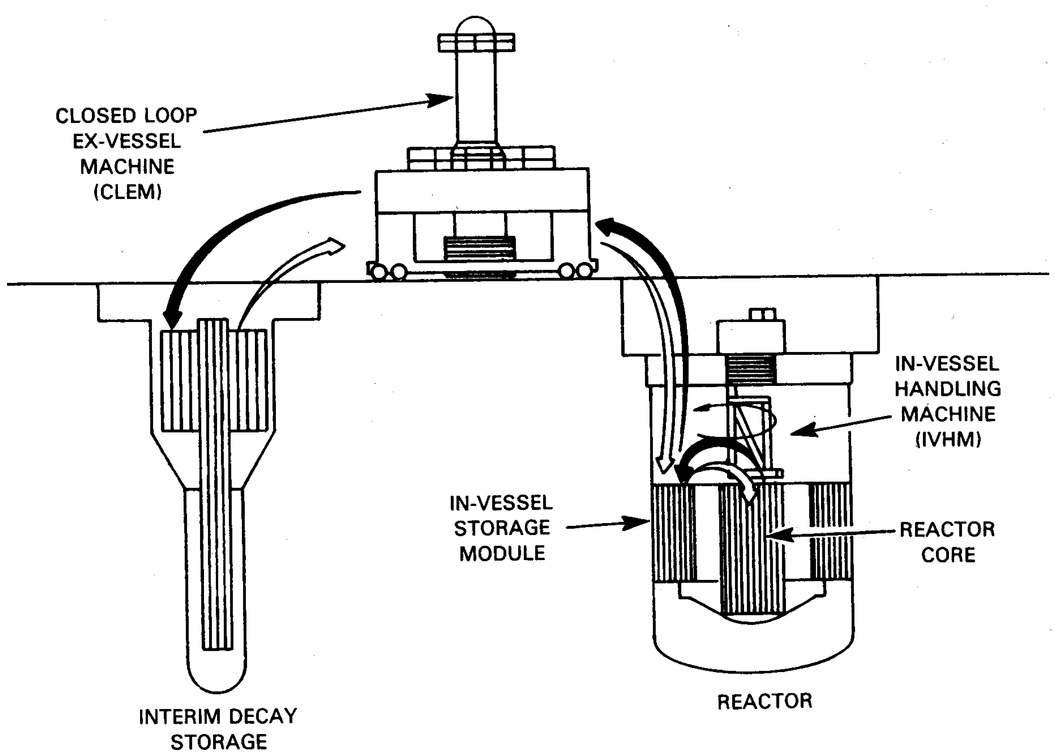

Fig. 5.5. FFTF driver fuel-handling sequence between interim decay storage and reactor vessel.

After reactor shutdown, the refueling plug in the reactor vessel lid is removed from the fuel transfer port and an adapter and floor valve are installed on the port. The control rod drive shafts are also disconnected, and the instrument trees are moved to their stored positions. The CLEM then picks up the CCP (with the fresh-fuel assembly) from the interim decay storage vessel, transports it to the reactor, and lowers it into the reactor through a fuel transfer port.

Inside the reactor vessel, the IVHM removes an SNF assembly from the reactor core and places it into an in-vessel storage position. The IVHM then removes the new assembly from the CCP and places the new assembly into the reactor core. Finally, the IVHM removes an SNF assembly from an in-vessel storage position and places it into the CCP.

The CLEM then removes the CCP with the SNF assembly from inside the reactor vessel and transfers the loaded CCP to the interim decay storage vessel. The SNF immediately after reactor shutdown generates significant quantities of decay heat. The SNF is transferred in a sodium-filled CCP to assure effective cooling of the hot SNF during transfer operations. After a suitable decay period, the bottom loading transfer cask picks up the SNF assembly from the interim decay storage vessel and transfers it out of containment to the long-term fuel-storage facility.

Before reactor startup, the IVHMs are placed in their stored positions in the reactor. The instrument trees and control rod shafts are restored to power-operation status, the adapter and floor valve are removed, and the fuel transfer port is sealed with its plug.

5.2.3 CRBRP Fuel Handling System

The CRBRP fuel-handling system (CRBRP, 1974a, 1974b, 1983) provides for replacement of fuel, blanket, control, reflector, and restraint assemblies. The system consists of the facilities and equipment needed to accomplish the normal scheduled refueling operations, as well as all other functions incident to handling of core components. Refueling operations involve transfer of core components between positions within the reactor vessel and between the reactor and the ex-vessel storage tank (EVST). Refueling operations can be accomplished only when the reactor is shut down.

A view of the general arrangement of the fuel-handling system is shown in Fig. 5.6. New fuel assemblies arrive at the plant in shielded and cushioned containers. They are unloaded and inspected in a shielded new fuel-handling facility located in the reactor service building (RSB). Upon acceptance of a core component, it is stored in a subcritical array in a storage facility located in the floor of the new fuel-handling facility.

In preparation for the refueling cycle, new fuel assemblies are removed from the storage facility in a shielded transfer machine and inserted into a gas-filled thimble in the EVST. During the course of this operation, the air atmosphere in the shielded transfer cask is exchanged for an inert, dry argon atmosphere, which is compatible with the liquid sodium environment in the EVST. The new core components are loaded into preheat tubes in the EVST at 475°F (246°C). After preheating, the EVTM is used to transfer the fresh fuel assemblies to sodium-filled core component pots (CCPs) that are also located in the EVST. There is one fuel assembly in each CCP. The process continues until a reactor-reload quantity of fuel has been accumulated for refueling.

After the reactor is shut down, the ex-vessel transfer machine (EVTM) transfers a new fuel assembly, in a sodium-filled CCP, from the EVST to the reactor through a 44.5' hatch in the reactor containment building (RCB) wall. When the new fuel assembly arrives at the reactor, it is discharged from the EVTM into a transfer position on the periphery of the reactor core in the reactor vessel.

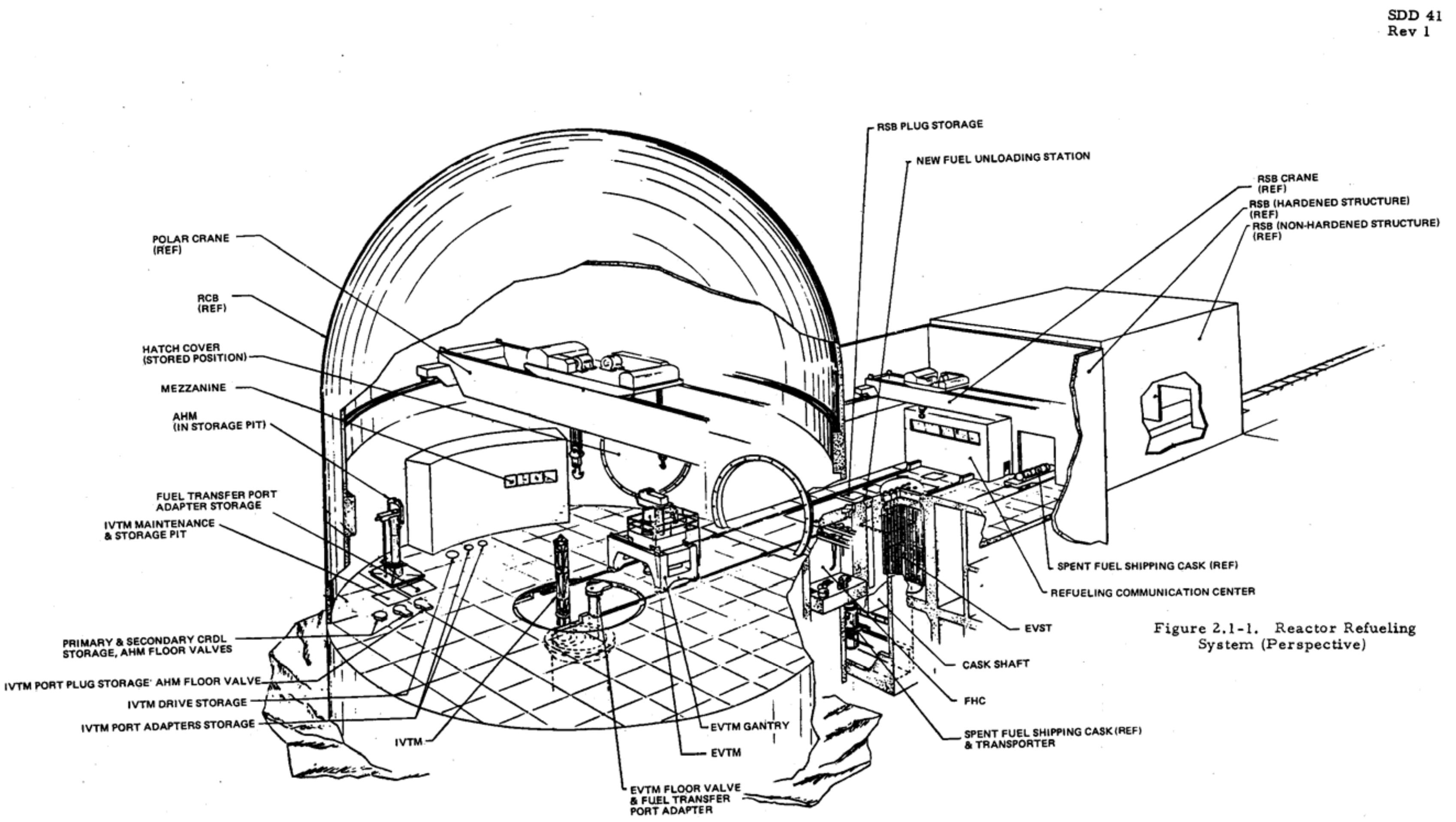

Fig. 5.6. CRBRP reactor refueling system.

At this point, the new core component is available for handling by the in-vessel transfer machine (IVTM). The IVTM withdraws an SNF assembly from its position in the reactor core and deposits it into an empty CCP. The IVTM then picks up a new fuel assembly from the CCP and inserts the assembly into position in the reactor core. Horizontal motion of the IVTM is accomplished by means of triple rotating plugs mounted in the reactor head (Fig. 5.7). By rotating these plugs in sequence, the IVTM, which is a simple straight-pull tubular device mounted on a port in the innermost rotating plug, can be indexed over any core or transfer position in the reactor.

When a CCP is full, the EVTM transfers the CCP with the SNF assembly in sodium from the reactor vessel to the EVST. After a suitable decay period, the SNF can be removed from the EVST and prepared for shipment from the plant.

5.3 LIQUID-SALT-COOLED REFUELING

Refueling differences exist between sodium-cooled reactors and the AHTR. For the AHTR, refueling temperatures are somewhat higher, the fuel geometry is different, the power density of the prismatic-block fuel-type SNF is 1 to 2 orders of magnitude lower, the vapor pressures of the liquid salts are much lower than those of sodium, and the liquid salt is transparent whereas the sodium is opaque. This section provides discussions of design considerations for the LS–VHTR fuel-handling system relative to sodium-cooled fast reactors.

5.3.1 Primary-System Refueling Temperatures

This survey of the EBR-II, FFTF, and CRBRP fuel-handling design, operation, and experience has shown that the temperature at which fuel handling was conducted in these systems was dictated by design choices. For example, in the EBR-II, fuel handling was conducted at 700°F (371°C), because this was the design bulk temperature for the primary tank environment (i.e., the normal operating reactor coolant inlet temperature). This temperature was selected to provide margin above the coolant melting temperature (98°C) in cold (peripheral, low-flow) locations within the large primary tank of the EBR-II. Fuel handling at this temperature is feasible because the structural materials used in EBR-II, primarily 304 and 316 stainless steels, have very good strength characteristics up to temperatures of 550°C and above. (The normal operating outlet temperature of the EBR-II is 473°C.)

In contrast, the loop-type FFTF primary-system design has a compact reactor vessel that does not have the large coolant volumes subject to stratification and flow stagnation like those of the EBR-II. Hence, the fuel-handling temperature is not based on prevention of coolant freezing, since this is less of an issue in the FFTF design, but rather upon maximum allowable fuel temperatures during refueling. This criterion arises because of the fuel-testing mission of the FFTF and the need to preserve test fuel integrity after irradiation. Figure 5.8 shows a thermal profile for a specified fuel assembly during FFTF fuel-handling operations (FFTF, 1983). Note that short-term assembly temperatures (cladding temperatures) as high as 538°C (1000°F) are allowed during fuel handling. (For comparison, the coolant temperature range in the FFTF during normal operation is 360 to 503°C.) The normal recommended fuel assembly refueling temperatures range is 204°C (400°F) to 249°C (480°F).

Based on liquid-sodium reactor design and operating experience, the process for determining the recommended fuel-handling temperature for the LS–VHTR should be based on assessments of the following: (1) thermal-hydraulic studies to determine the minimum acceptable temperature margin above the coolant melting temperature to maintain natural circulation coolant flow during refueling, (2) stress analysis of metallic structure temperatures (both reactor vessel structure and in-vessel fuel-handling machine structure) that are required to maintain acceptable margins below material design-basis limits, and (3) thermal analysis of the fuel-element refueling temperatures to determine acceptable margins below the fuel temperature limit.

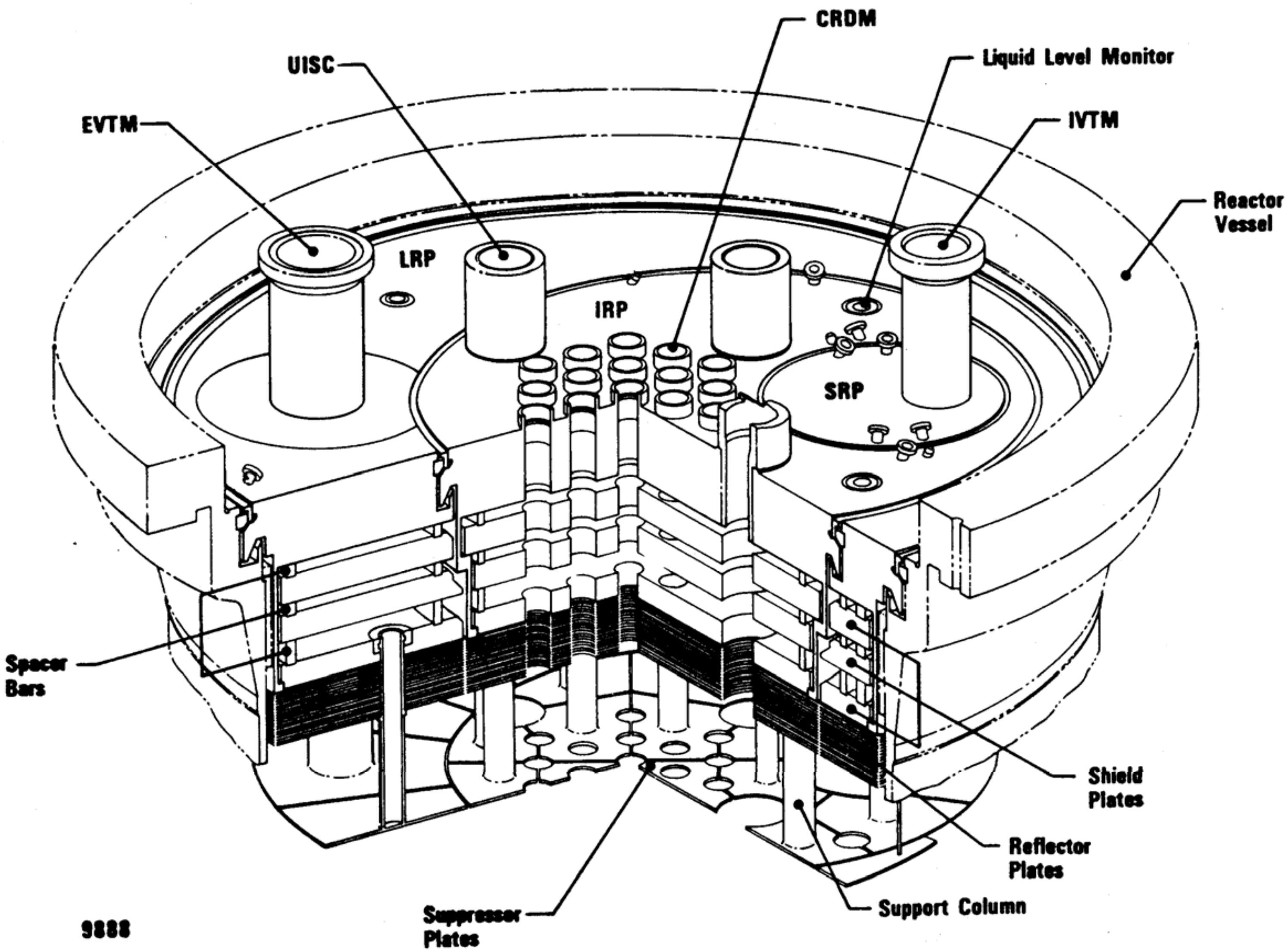

Fig. 5. 7. CRBRP reactor vessel head rotating plugs.

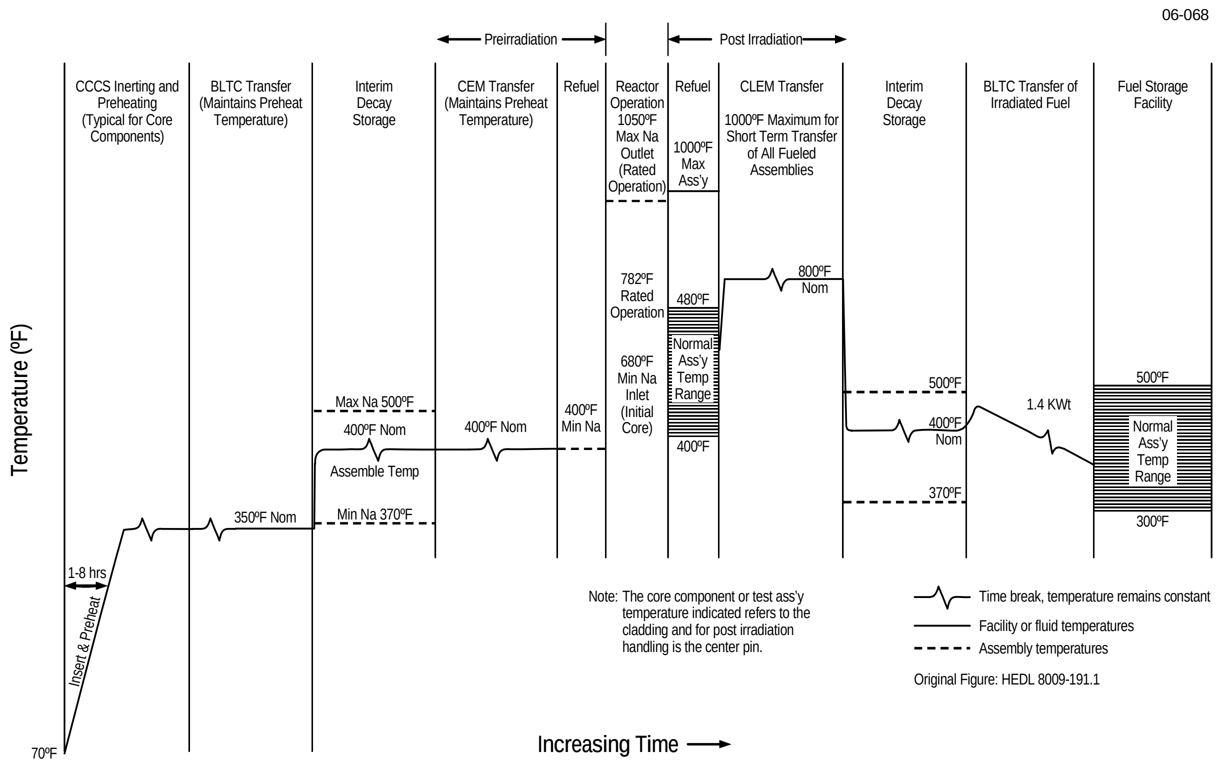

Fig. 5.8. Thermal profile for FFTF fuel and test assembly handling.

5.3.2 In-Vessel Fuel-Handling Equipment Design and Fuel Restraint Design

If it is assumed that the in-vessel fuel-handling machine can be constructed of the same advanced alloy as the reactor vessel (750°C operating temperature), then it is reasonable to expect that the LS–VHTR design of the in-vessel fuel-handling machine could closely resemble the design proposed for the GT-MHR machine (General Atomics, 1996). This machine is very similar to the Fort St. Vrain machine (Sect. 4.1.2) except the design has been updated to reflect (1) improvements in selected components (such as instrumentation and control) and (2) what was learned in Fort St. Vrain operations.

5.3.3 Design Requirements Relevant to the LS-VHTR

Many of the design requirements for the EBR-II, FFTF, and CRBRP fuel handling systems have relevance to the proposed prismatic-fuel LS-VHTR design and operating characteristics. Some elements are also relevant to the online refueling systems for the pebble bed and stringer fuel designs, and for reflector and moderator replacement in these designs.

- All in-vessel fuel-handling operations are conducted remotely at high temperatures.

- Fuel-handling mechanisms are designed to operate for extended periods with very high reliability. The mechanisms are repairable and replaceable, have low failure rates, and sustain failures only at very long time intervals. Mechanisms are constructed from materials (mostly stainless steel) that are appropriate for the mission requirements.

- Fuel-handling mechanisms are designed for precise alignment that can be maintained for prolonged periods of operation, since remote realignment is difficult while the equipment is submerged. Note that this requirement is more critical for a sodium reactor with an opaque coolant than it is for a liquid salt cooled reactor where optical techniques can be used to monitor and guide the refueling process (see Sect. 6).

- Positioning of grippers for fuel elements is performed with automated computer control; components are designed with guides (funnels, slides, ports) to facilitate mating and gripping operations.

- To the extent possible, sensing mechanisms are employed to verify all positioning, grasping, movement, transfer, and release functions. Such mechanisms include instrumentation on rotating and elevating machinery, as well as manual "feel" through mechanical links.

- Fuel-handling and transfer equipment is designed to handle one subassembly at a time.

- Grasping and gripping mechanisms are designed to minimize the potential for dropping a subassembly.

- Transfer operations are accomplished with positive, mechanically-actuated displacement actions (without reliance on gravity).

- In-vessel fuel handling-mechanisms are designed to have drive mechanisms with vertical and rotary motion.

- For the EBR-II, shaft penetrations through the primary tank cover are equipped with packing-gland seals to minimize escape of argon cover gas and prevent air in-leakage. For the FFTF and the CRBRP, vessel head ports are mated to special removable valves to control leakage. Similar methods will be required for salt-cooled reactors.

- Wherever possible, fuel-handling mechanisms are designed with additional capability to accommodate nonstandard operations with bent, damaged, or incorrectly positioned subassemblies.

- To minimize reactor shutdown time, the overall system configuration is selected on the basis of time-motion studies and the refueling interval requirement. System configuration studies include tradeoff studies such as leaving parts of the refueling machine in the vessel during operation (FFTF) versus full removal of the refueling equipment (CRBR).

- Inspection and maintenance must be considered throughout the design. One of the major conclusions from the operation of large fast reactors is that inspection requirements often have major negative impacts on plant availability.

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6. INSTRUMENTATION AND CONTROL

Associated with refueling, inspection and maintenance (RIM) operations is the need for sensors. In addition to the traditional sensors used in other high-temperature reactors (thermocouples, ultrasonic sensors, etc.), liquid salts allow the use of optical systems. Salts are transparent between 200 and 2500 nm (50,000 to 4000 wave numbers). This includes the UV, visible, and near-infrared, with some transparency into the infrared from 2500 to 5000 nm (4000 to 2000 wave numbers). In other words, these salts are transparent over a wider range of the spectrum than is water. The liquid salt, depending upon its composition and activation, provides some radiation shielding for the optical systems. The coolant properties create new sensor options.

Laser signals into the reactor vessel and return signals would be transmitted by the use of mirrors placed either in a periscope with a diamond or sapphire window or directly in the liquid salt (Fig. 6.1). Polished noble-metal mirrors can be immersed in the salt. Because the salt is a fluxing agent, the mirrors would be expected to remain clean (something that would not occur in water or in most other fluids). The signal-to-noise ratio of a laser or other light signal can be maximized by (1) choosing the frequency of the light, (2) using polarized light, and (3) adjusting the power level of the laser.

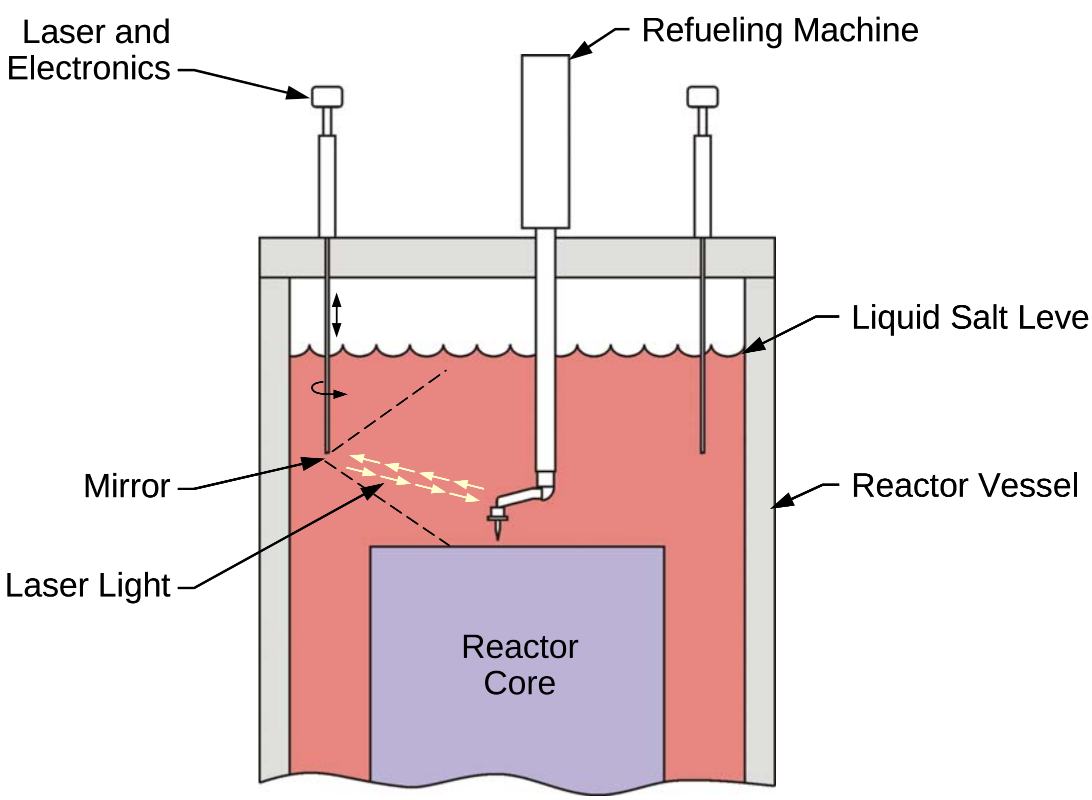

Fig. 6.1. Mirror system for transmitting signals into and out of the reactor vessel.

6.1 PRECISION METROLOGY

Precision metrology is the use of multiple laser range finders at defined locations with appropriate software systems to map three-dimensional environments. It is used today to map everything from chemical refineries to railroad right-of-way clearances to the inside of fusion energy machines (Kugel et al. 2001; Menon and Slotwinski, 2004). Multiple lasers at well-defined locations measure the direction and range to each object. The laser frequency and power levels are chosen by the designer or, in high-end systems, selected by the operator. There is a massive industrial experience base and multiple commercial suppliers for a variety of applications. However, the technology has not been developed for liquid-salt systems and development would be required for this specific application. Frequencies for the LS-VHTR would be selected for the best transmission of light through the salt where there is not significant light emissions from the reactor systems (such as from thermal radiation). Polarized light can be used to further boost the signal to noise ratio. Software programs create a three-dimensional model of the environment.

For systems measuring a few tens of meters in scale, experience in non-salt systems indicates that visible surfaces can be mapped to a resolution as small as 0.1 mm (Fig. 6.2 shows an example image of a dime obtained by laser scanning of the surface from a distance of 4.2 meters). The technology in a commercial form has become available only within the last decade, with the availability of low-cost computers that allow rapid creation of three-dimensional images with a precision of ~5 mm. Special systems are designed for much higher resolution and more challenging conditions. Systems for fusion experiments are being designed and tested for radiation levels of 10^6 rad/h under vacuum conditions and 6-T magnetic fields. The lasers and other equipment are located outside the hostile environment.

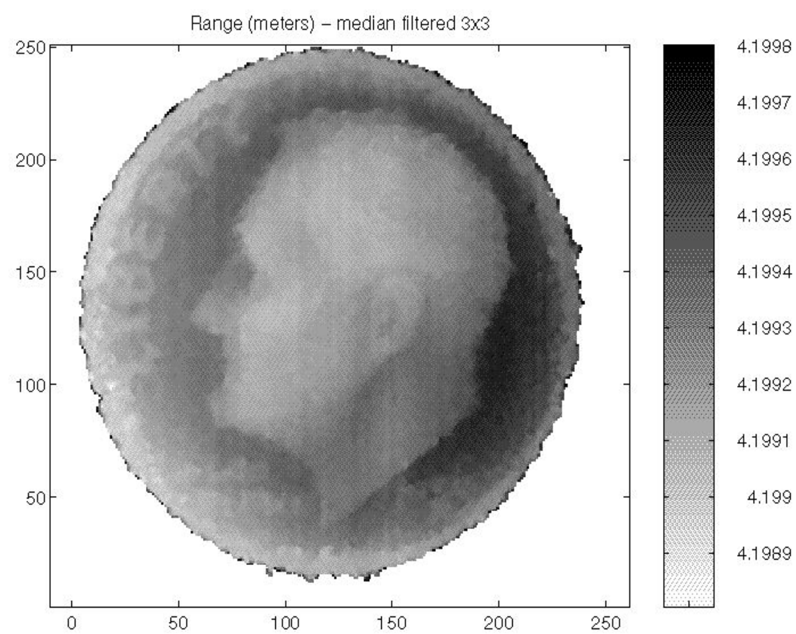

Fig. 6.2. Image of a dime obtained by laser scanning of the surface from a distance of 4.2 m.

Precision metrology has multiple applications for a LS-VHTR.

- *Refueling.* Precision metrology will allow operators to see operations and provide the refueling machine with information where everything is located. Modern robotic systems, including refueling machines, have control systems to prevent collisions with solid objects. Precision metrology creates three-dimensional maps to provide the input to those systems.

- *Vibration analysis*. The system detects component vibrations that may indicate potential problems.

- *Fluid flow*. If small nanoparticles are present in the fluid, the liquid-salt flow velocities can be measured. This technique is used in the laboratory and in parts of the chemical industry.

- *Inspection*. Regular scans of the reactor interior can confirm that no significant unplanned changes in geometry have occurred. This includes finding loose parts and conducting inspections to see that no significant changes in surfaces—the same application as used in fusion devices. The high-precision systems being developed for fusion applications are designed to find cracks and other damage on the interior surfaces of fusion machines, as shown in Fig. 6.3.

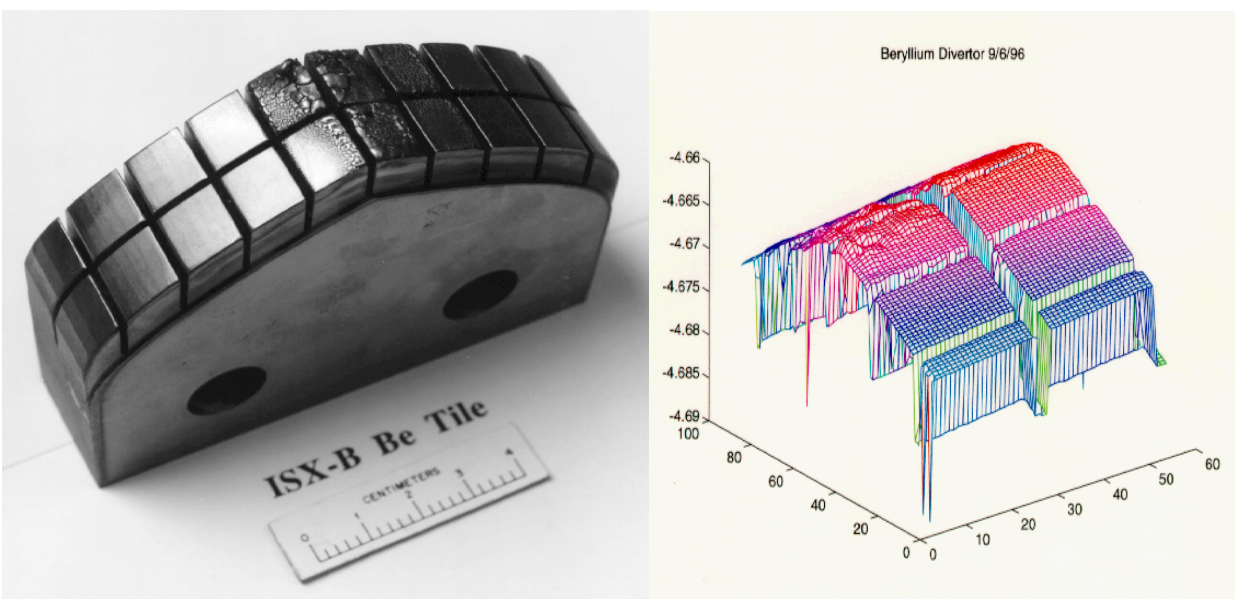

Fig. 6.3. Precision metrology surface inspection of the first-wall component of a fusion reactor.

Precision metrology is applicable for all operations when the reactor is shut down. It may also be applicable for monitoring during power operations; however, significant additional challenges remain. The liquid-salt coolant provides radiation shielding for the system optics (mirrors) and other components. Online monitoring, if proven feasible, would provide an extraordinary and unique diagnostic tool.

6.2 SPECTROSCOPY

Spectroscopy is the measurement of light intensity versus frequency. Remote high-temperature measurement systems often use some form of spectroscopy to measure temperatures. In the laboratory, high-temperature salt properties (purity, composition, etc.) can be measured by spectroscopy. This is a standard technique used in the chemical industry for online monitoring of the chemical composition of flowing streams in chemical plants. Properties that can be measured include the following.

- *Temperature*. Remote high-temperature measurement systems traditionally use some form of spectroscopy to determine temperature.

- *Salt purity, density, chemical composition, and other properties*. In the laboratory, high-temperature salt properties are measured by spectroscopy. Laser or other light is sent through the salt, and the transmission of the light is measured as a function of frequency. In more sophisticated systems, secondary emission lines are measured. Salt impurities that can be measured to very low concentrations include uranium, the actinides, iron, chromium, and nickel. The chemical valence state can also be measured. This is likely to be the preferred method for monitoring the concentration of impurities and the redox potential of the salt and thus the performance of the salt cleanup systems. It would be the equivalent of the instrumentation used to monitor water chemistry in an LWR.

- *Radiation decay*. In some salts, neutron activation of the salt leads to a radioactive decay process that creates high-energy electrons. As the electrons in the salt slow down, photons are emitted that peak in the blue part of the visible spectrum. Because this part of the spectrum is far from the thermal infrared signal caused by the high temperatures, it will be detectable. Depending upon half-lives of the activated nuclides and flow velocities, this phenomenon may allow mapping of salt flow patterns above the reactor core.

6.3 TELEVISION

TV cameras have been developed for a wide variety of hostile environments and are applicable to refueling. They represent an alternative optical method but will not provide the clarity of view of precision metrology. Specialized TV cameras such as fiberscopes have been developed with high radiation resistance and enable the electronics to be positioned in cooled low-radiation zones.

7. CONCLUSIONS

A review of historical experience with other types of high-temperature reactors provides confidence that refueling of an LS-VHTR is practical and can be accomplished in a reasonable period of time. Examples of mechanical refueling solutions with similar temperatures, fluids, and fluid interface conditions have been presented. Three principal options for refueling three basic types of reactor cores (prismatic, pebble, and stringer fuel geometries) have been identified. However, the development of the refueling machinery for this reactor, with its relatively high refueling temperatures, will require a major development effort. Trade studies necessary for the selection of the fuel geometry will require consideration of reactor core behavior, fuel fabrication, and online versus off-line refueling. Materials issues must be addressed for a wide variety of conditions that occur during refueling. Evaluations of temperatures of SNF assemblies during refueling normal and off-normal operations will be required. The transparency of the salt may allow advanced instrumentation methods that will greatly simplify RIM operations relative to those for sodium-cooled and other high-temperature reactors.

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8. REFERENCES

Althaus, D., and N. Brahy. 1987. Refueling System Fabrication and Testing," *Nuclear Technology,* **78**, 284–294 (September).

Cabell, C. P. 1980, *A Summary Description of the Fast Flux Test Facility*, HEDL-400, Hanford Engineering Development Laboratory, December.

Cahalan, J. E., and T. A. Taiwo, 2006, *Liquid Salt–Very High-Temperature Reactor: Survey of Sodium-Cooled Fast Reactor Fuel Handling Systems for Relevant Design and Operating Characteristics*, ANL-GenIV-069, Argonne National Laboratory, Argonne, Illinois, March 31.

Cornell, R. M., N. F. Haines, and S. B. Fisher, 1995, "Facilities and Techniques for On-Site and Off-Site Inspection of Irradiated Fuel Elements and Components from Nuclear Electric's Advanced Gas-Cooled Reactors," Paper 54, *Proceedings of the International Conference on Fuel Management and Handling, Edinburgh, March 20–22, 1995*, British Nuclear Energy Society, London.

CRBRP, 1974a, *Clinch River Breeder Reactor Plant Reference Design Report, Vol. 1*, June.

CRBRP, 1974b, *Clinch River Breeder Reactor Project: 1974 Technical Progress Report.*

CRBRP, 1983, *Clinch River Breeder Reactor Plant System Design Description, Reactor Refueling System*, SDD-41, Rev. 27, July.

Dixon, G., and N. Penny, 1995, "Underwater Fuel Inspection Capabilities at AEA Technology, Windscale," Paper 55, *Proc. of the Int. Conf. Fuel Management and Handling, Edinburgh, March 20–22, 1995*, British Nuclear Energy Society, London.

deZwaan, S. J. 2005, *The Liquid Salt Pebble Bed Reactor: A New High-Temperature Nuclear Reactor*, Department of Radiation, Radionuclides and Reactors, Delft University of Technology, Delft, The Netherlands, November.

EBR-II, 1971, *EBR-II System Design Descriptions*, *Volume II, Chapter 5*: "Fuel Handling System," Argonne National Laboratory, Argonne, Illinois, June 15.

FFTF, 1983, *Fast Flux Test Facility System Design Description, No. 41, Part 1*: "Reactor Refueling System," Rev. 11, April 6.

Forsberg, C. W. 2006a, "Alternative Passive Decay-Heat Systems for the Advanced High-Temperature Reactor," Paper 6055 (CD-ROM), *International Congress on Advanced Nuclear Power Plants, June 4–8, 2006, Reno, Nevada*.

Forsberg, C. W. 2006b, "Goals, Requirements, and Design Implications for the Advanced High-Temperature Reactor," ICONE14-89305 (CD-ROM), *14th International Conference on Nuclear Energy, American Society of Mechanical Engineers, Miami, Florida, July 17-20,* American Society of Mechanical Engineers.

Forsberg, C. W. 2006c, “Advanced-High-Temperature-Reactor Spent-Fuel Characteristics and Repository Impacts,” (CD-ROM), *2006 International High-Level Radioactive Waste Management Conference, April 30–May4, 2006, Las Vegas, Nevada*.

Forsberg, C. W. 2006d, “Developments in Molten Salt and Liquid-Salt-Cooled Reactors, Paper 6292 (CD-ROM), *International Congress on Advanced Nuclear Power Plants, June 4–8, 2006, Reno, Nevada*.

General Atomics, 1996, *Gas Turbine-Modular Helium Reactor (GT-MHR) Conceptual Design Description Report*, General Atomics Report 910720, San Diego, California, Rev. 1, July.

Ingersoll, D. T., and C. W. Forsberg, 2006, “Overview and Status of the Advanced High-Temperature Reactor,” Paper 6264 (CD-ROM), *International Congress on Advanced Nuclear Power Plants, June 4–8, 2006, Reno, Nevada*.

Ingersoll, D. T., et al. 2005, *Status of Physics and Safety Analyses for the Liquid-Salt-Cooled Very High-Temperature Reactor (LS-VHTR)*, ORNL/TM-2005/218, Oak Ridge National Laboratory, Oak Ridge Tennessee, December.

Kim, T. K., T. A. Taiwo, and W. S. Yang, 2005, *Preliminary Neutronic Studies for the Liquid-Salt-Cooled Very High Temperature Reactor (LS-VHTR)*, ANL-GenIV-052, Argonne National Laboratory, Argonne, Illinois, August 31.

Koch, L. J., *Experimental Breeder Reactor-II (EBR-II): An Integrated Experimental Fast Reactor Power Station*, Argonne National Laboratory, Argonne, Illinois.

Kugel, H. W., D. Loesser, A. L. Roquemore, M. M. Menon, and R. E. Barry, 2001, “Precision Metrology of NSTX Surfaces Using Coherent Laser Radar Ranging,” *Review of Scientific Instruments*, **72**(1), 533–536 (January).

Menon, M. M., and A. Slotwinski, 2004, “Novel Doppler Laser Radar for Diagnostics in Fusion Reactors,” *Review of Scientific Instruments*, **75,** 4100–4102.

Mottershead, K. J., D. W. Beardsmore, and G. Money, 1995, “Dropped Fuel Damage Prediction Techniques and the DROFU Code,” Paper 43, *Proceedings of the International Conference on Fuel Management and Handling, Edinburgh, March 20–22, 1995*, British Nuclear Energy Society, London.

*Nuclear Applications and Technology (*now *Nuclear Technology),* 1970, **8**(2) (Entire issue).

Paget, J. A., 1967, *A Survey of Technical Considerations Applicable to an On-Line Refueling System for Large HTGRs*, GA-8075, General Atomics, San Diego, California, June 30.

Peterson, P. F., and H. Zhao, 2006, “A Flexible Baseline Design for the Advanced High Temperature Reactor Using Metallic Internals (AHTR-MI),” Paper 6086 (CD-ROM), *International Congress on Advanced Nuclear Power Plants, June 4–8, 2006, Reno, Nevada*.

Romrell, D. M., D. M. Art, R. D. Redekopp, J. B. Waldo, and J. L. Marshall, 1989, “Fast Flux Test Facility Fuel Handling Experience—November 1979 to August 1988,” *Nuclear Technology*, **86**, 264–274 (September).

Rosenthal, M. W., P. N. Haubenreich, and R. B. Briggs, 1972, The Development Status of Molten Salt Breeder Reactors, ORNL-4812, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Smith, P. G. 1961. "High-Temperature Molten-Salt Lubricated Hydrodynamic Journal Bearings," *ASLE Transactions*, **4**, 263–274.

Thoma, R. E. 1971. *Chemical Aspects of MSRE Operations*, ORNL-4658, Oak Ridge National Laboratory, Oak Ridge, Tennessee, December

U.S. Nuclear Regulatory Commission, 1991, *Fort St. Vrain Final Safety Analysis Report*, Rev. 9, Washington, D.C., July 22.

Williams, D. F. 2006, "Salt Selection of the LS-VHTR," Paper 6160 (CD-ROM), *International Congress on Advanced Nuclear Power Plants, June 4–8, 2006, Reno, Nevada.*

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APPENDIX A: FFTF FUEL-HANDLING SYSTEM

This appendix provides a brief description of the Fast Flux Test Facility (FFTF) fuel handling system and its operation as described by Cabell (1980) and in the *Fast Flux Test Facility System Design Description* (FFTF, 1983). The description is limited to those system features that are potentially relevant to the refueling of a liquid-salt very high-temperature reactor (LS-VHTR). Because the FFTF was designed as a reactor to test fuel, it has additional capabilities and equipment compared with a sodium-cooled fast reactor designed only to produce electricity.

A.1 FFTF FUEL-HANDLING SYSTEM

The FFTF refueling system includes facilities for the receipt, conditioning, storage, installation and removal of all core components from the reactor core (driver fuel assemblies, control assemblies, and test assemblies) that are routinely removable. Limited examination and packaging capabilities are also provided. All principal equipment incorporates an inert argon gas environment for the fuel and test assemblies. Refueling system components are located both inside and outside the reactor. An overview of the FFTF reactor refueling facilities is shown in Fig. A.1. The systems that make up this facility can be divided into three groups.

- *Ex-reactor support systems*. A variety of specialized systems are required to receive, inspect, and prepare fresh fuel for the reactor and to inspect, prepare, and ship spent nuclear fuel (SNF) from the reactor. These include two core component conditioning stations, a test assembly conditioning station, a bottom-loading transfer cask, an interim decay storage vessel, an interim examination and maintenance cell, a fuel storage facility, and a cask load-out station. Most of these systems can be operated while the reactor is operating and during refueling operations.

- *Transfer system*. The principal system used to transfer fuel and other components to and from the reactor vessel is the closed loop ex-vessel machine (CLEM) shown in Fig. A.2. The CLEM loads all components into the reactor vessel and removes all components from the reactor vessel and operates only when the reactor is shutdown. Under most conditions, CLEM moves a core component pot (CCP) that contains the fresh or SNF fuel to and from the reactor vessel. In a fast reactor, the core power density is very high; thus, there is significant decay heat in each SNF assembly immediately after reactor shutdown. To prevent fuel failure from overheating, the SNF is kept in sodium at all times to ensure effective cooling. This is accomplished by transferring each fuel assembly in its own pot of sodium—the CCP. CLEM is also used to transfer a variety of other components within the reactor containment.

- *Reactor refueling systems*. A series of systems associated with the reactor and reactor vessel are required for refueling. Outside the reactor vessel these include three fuel transfer ports, eight test position spool pieces, two floor valve adapters and the test transfer port, and the reactor containment building cranes. Inside the reactor vessel are three in-vessel handling machines (IVHMs) and three in-vessel storage modules. Three IVHMs are required because of the mission of the FFTF that requires closed test loops in the reactor core which interfere with direct access to the entire core with one machine.

FFTF was designed for a cycle of 102 days of operations and 28 days for refueling. In a refueling outage, typically one-third of the reactor core would be replaced. This included 24 driver fuel assemblies, 6 control rods, 12 reflector assemblies, and 2 closed-loop test assemblies.

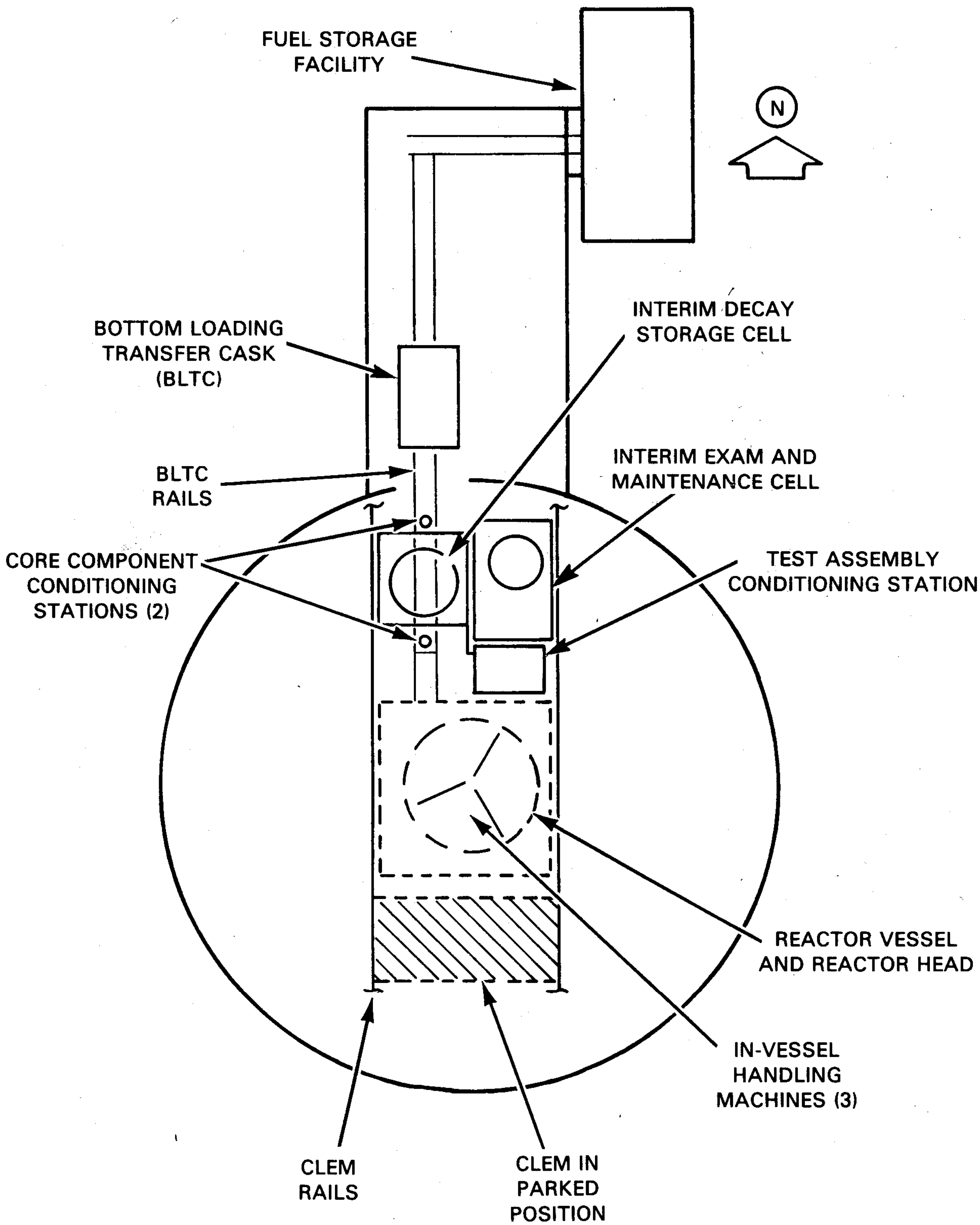

Fig. A.1. General arrangement of reactor refueling facilities.

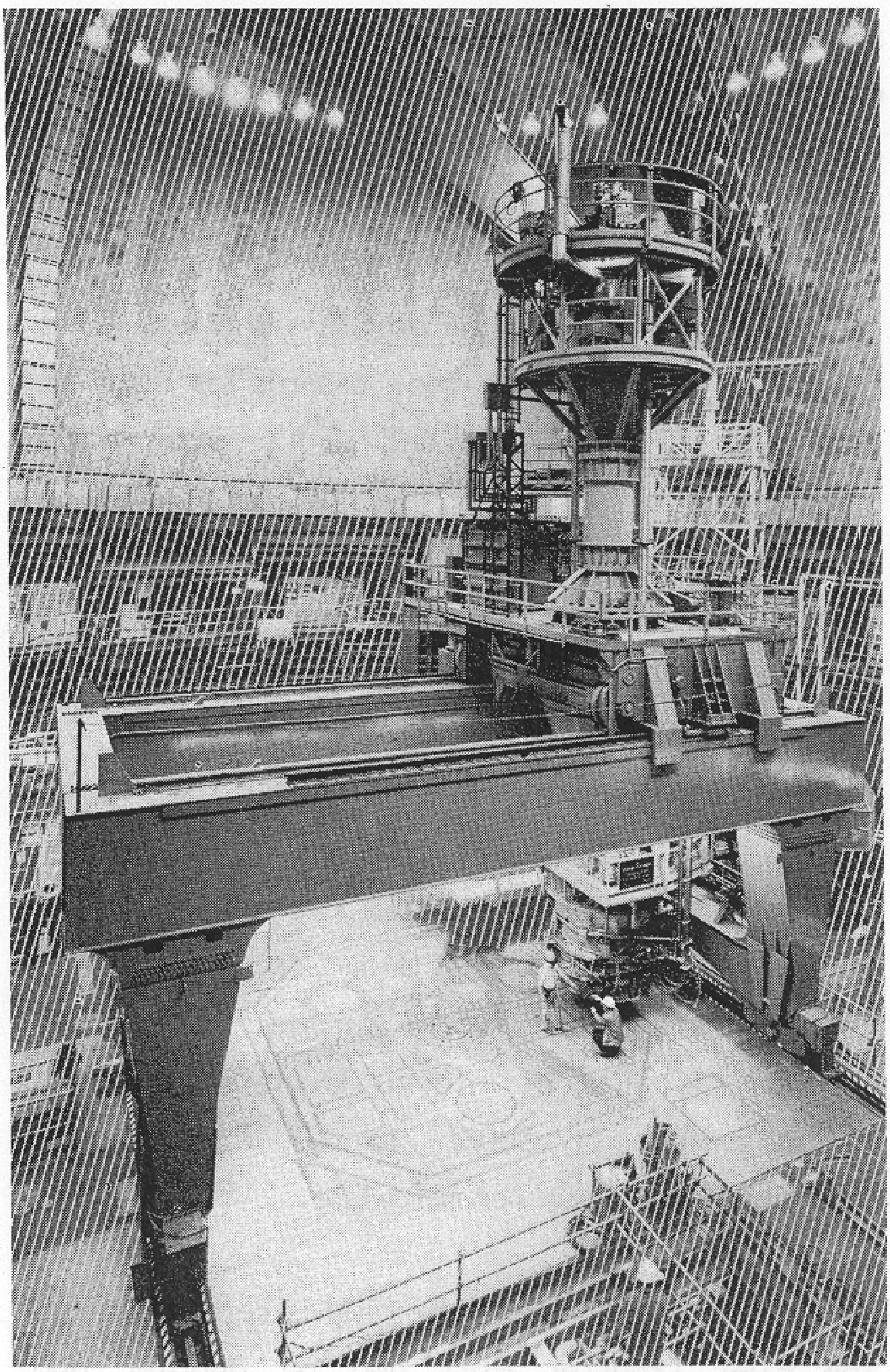

Fig. A.2. Closed loop ex-vessel machine (non-shaded component).

A.2 SYSTEM FUNCTIONS

The reactor refueling system handles three types of core assemblies: 12-ft assemblies, such as driver fuel; 40-ft assemblies, such as fuels open test assemblies, which are loaded into the reactor full-length but are cut into two sections before removal; and 40-ft assemblies, such as materials open-test assemblies, which are loaded into and removed from the reactor as full-length sections. The typical sequence for driver fuel-handling operation is summarized in Sect. A.2.1, and a brief description of contamination control and inventory control practices is provided in Sect. A.2.2. The FFTF refueling systems are more complex than those of a power reactor because the FFTF is a fuel testing machine that must handle many special test fuel assemblies, many of which require special handling.

A.2.1 Fuel Handling

Fuel handling involves a series of complex operations (Figs. A.3 and A.4). After receipt and unpacking of the fresh fuel assemblies from the fuel fabrication plant, each 12-ft fuel assembly is lowered by the crane into one of the two core component conditioning stations. In this station, the assembly is flooded with argon and heated to ~232°C (450ºF). The assembly is then picked up by the bottom-loading transfer cask and is transported, in an argon atmosphere, to the interim decay storage vessel. The assembly is lowered into a sodium-filled CCP in the interim decay storage vessel, where it is held in liquid sodium at 260−316°C (500−600°F) until the reactor is shut down for refueling.

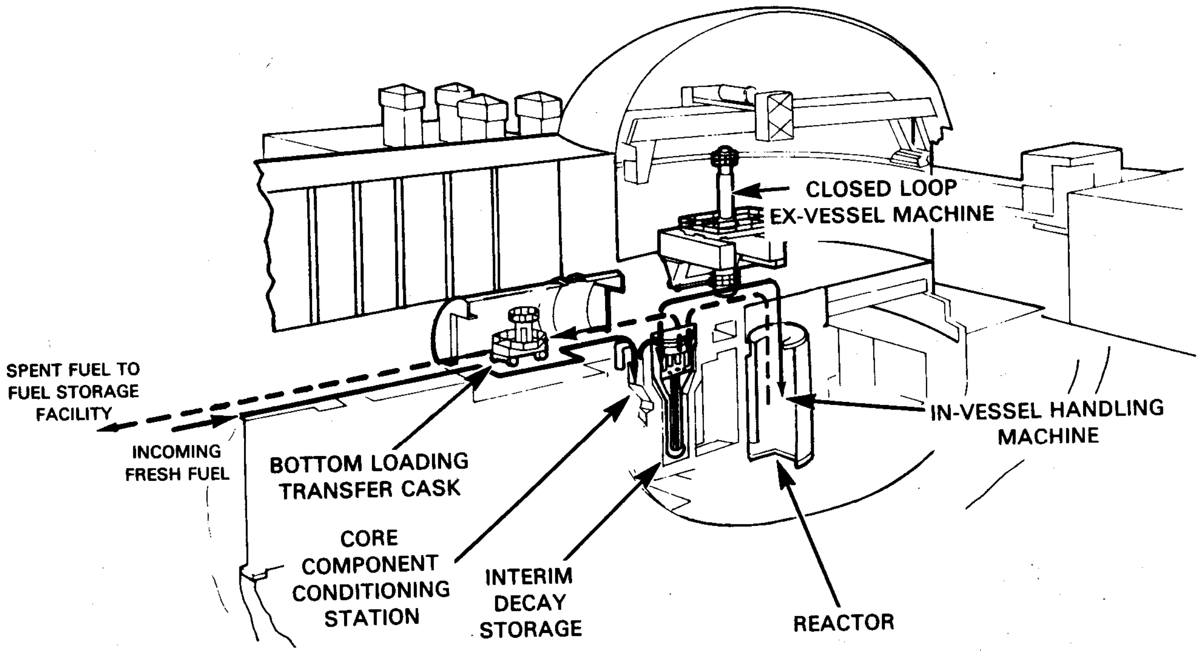

Fig. A.3. Driver fuel−handling sequence.

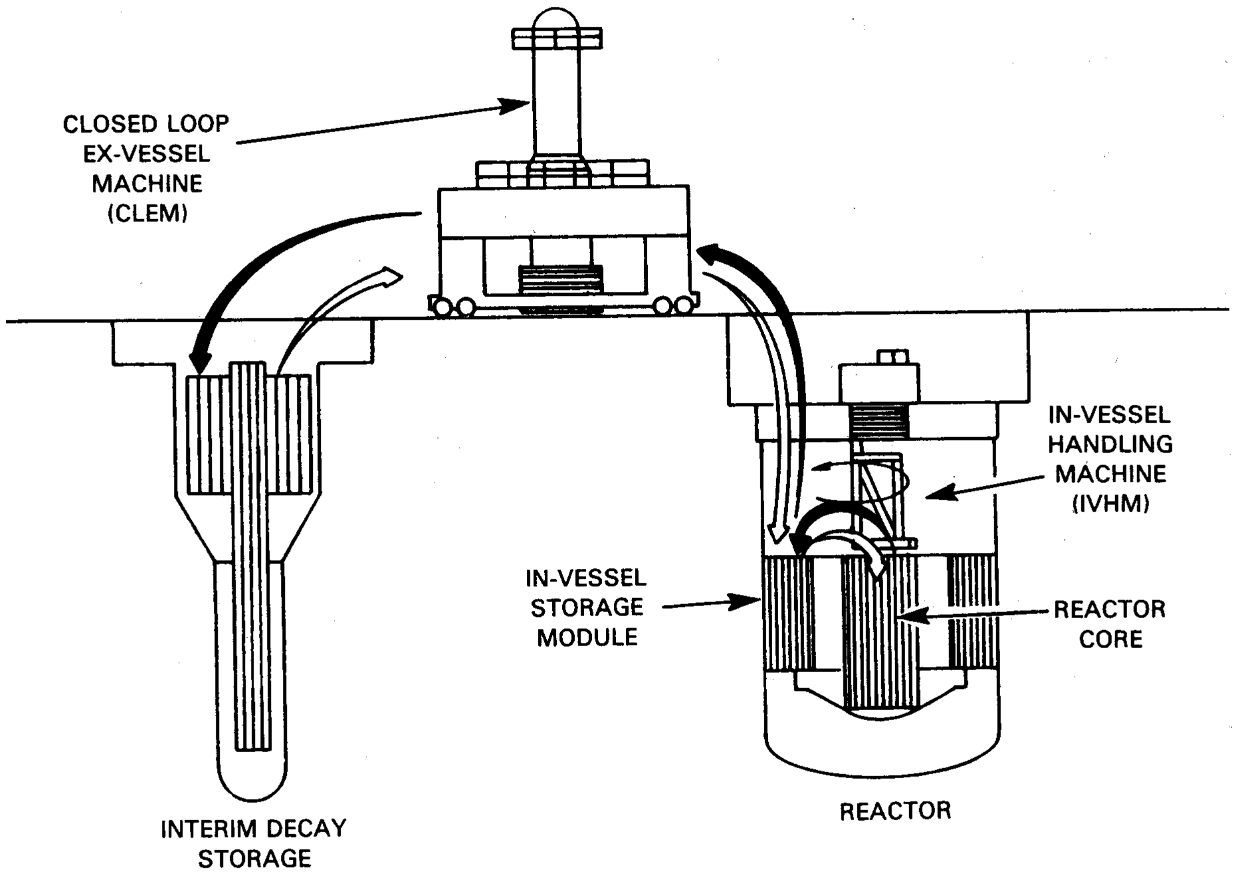

Fig. A.4. Driver fuel-handling sequence between interim decay storage and reactor vessel.

After the reactor has been shut down, plugs are removed from the fuel transfer ports on the reactor vessel lid and adapters and floor valves are installed on the ports. The control rod drive shafts are also disconnected, and the instrument trees are moved to their stored positions. The CLEM then picks up the CCP containing the fresh fuel assembly from the interim decay storage vessel, moves it to above the reactor vessel, and lowers it into the reactor vessel through a fuel transfer port.

In the reactor vessel, the IVHM removes the SNF assembly from the core and places it into an in-vessel storage module position. The IVHM then removes the new assembly from the CCP and places it into the core. Finally, the IVHM removes the SNF assembly from the in-vessel storage module and places it into the empty CCP.

The CLEM then picks up the CCP, containing the SNF assembly, and transfers it to the interim decay storage vessel. The cyclic process of moving fresh fuel to the reactor core and SNF back to the interim decay storage vessel continues until all the refueling operations have been completed. Before reactor startup, the IVHMs are placed in their "stored" positions in the reactor, the instrument trees and control rod shafts are restored to power-operation status, the adapters and floor valves are removed, and the fuel transfer ports are sealed with their plugs.

After a suitable decay period, the bottom-loading transfer cask picks up the SNF assemblies from the interim decay storage vessel and transfers them out of containment to the fuel storage facility. This operation can be done after the reactor is back online.

CLEM is also used to transfer a variety of test assemblies and other components into and out of the reactor. A variety of special transfer techniques are used for these other components.

A.2.2 Vessel Containment System

The confinement of fission gases and contaminants, avoidance of sodium vapors in the containment building, and avoidance of air with moisture in the gas space above the sodium are of critical importance during all phases of fuel handling. Gaining access to all the inerted or potentially contaminated facilities requires the use of equipment such as floor valves (Fig. A.5) and adapters that are unique to the particular facility or port involved. The most important adapters are the two floor valve adapters and the test transfer port.

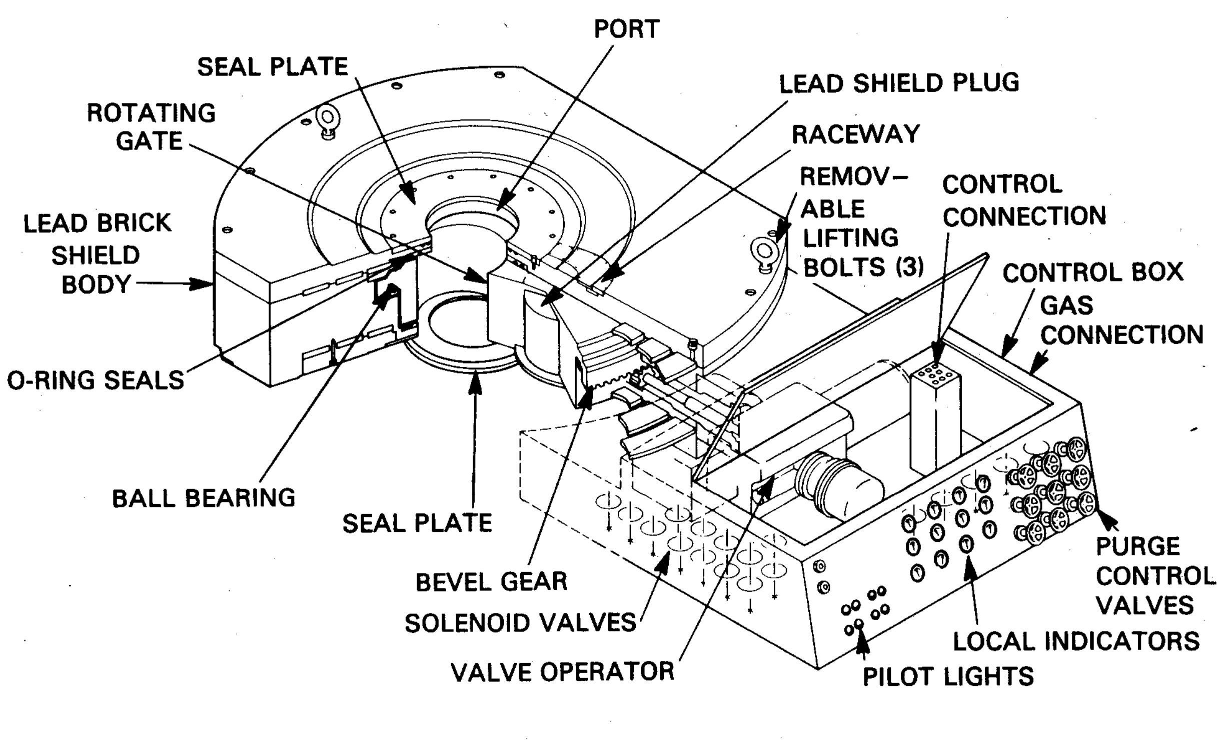

Fig. A.5. Floor valve.

A.2.3 Inventory Control

Because of the number and various types of assemblies being handled, as well as the inability to directly observe the assemblies at most residence or transfer points, an inventory-control system is put in place. This tracks all core assemblies, CCPs, and associated components by location and also records the status of all storage locations.

A.3 SYSTEM COMPONENT DESCRIPTIONS

A.3.1 Closed Loop Ex-Vessel Machine (CLEM)

The function of CLEM is to transfer reactor core components and test assemblies (which may be up to 40 ft long) between the various FFTF refueling stations. The SNF is always transferred in CCPs to assure cooling of the SNF and avoid overheating the SNF. CCPs may or may not be used for other components.

The CLEM is 61 ft high, weighs 217 tons, and moves on a 40-ft-wide transporter on tracks in the operating deck (Fig. A.6). The CLEM has a 10-kW(t) heat removal capability and handles one component at a time. The temperature of the CLEM is controlled by a cold-wall cooling system and an electrical heating system. The cold-wall cooling system consists of an 8-in. pipe having an array of straight fins attached to the outside. The normal cooling mode for the CLEM is by internal radiation of heat from the outer wall of the sodium-cooled CCP (with SNF assembly) to the cold wall, which is externally cooled by ambient air circulated over the fins. During normal core component handling operations, the fuel pin cladding temperature of the driver fuel is limited to 538°C (1000ºF).

The CLEM operator controls all machine operations from the control console in the cab mounted on the transporter trolley. The control of CLEM is semiautomatic; that is, operations are performed automatically after the operator initiates a command. Interlocks to control the machine's operations have been incorporated into the design to meet safety requirements. Manual overrides are also provided to permit the operator to take control if necessary. The grapple speed and push-pull load requirements in handling fuel are controlled through all modes of operation. Indexing of the CLEM to a refueling station is accomplished by manually operated controls.

The CLEM is hermetically sealed to a refueling station by lowering the movable closure valve, which mates with a floor valve. Sealing this interface is accomplished by two inflatable seals (providing a double barrier), which engage the floor valve. The air trapped between the movable closure valve and floor valve is purged by the plant argon supply and vent system through hose connections made after the CLEM is indexed to a fuel transfer port or test transfer port.

Electrical and instrumentation connections are also made at each refueling station to control and monitor the operation of the floor valve and provide the interlock connections to ensure safe operation (e.g., CLEM and the IVHM are interlocked at the reactor to prevent simultaneous operation in their common operating space).

The major subassemblies of the CLEM include the grapple drive system, cask body modules, service platforms, a movable closure valve that mates with the floor valve, the cold-wall assembly, and a seal monitoring system. It also has a variety of special features for other required test-reactor operations such as moving test assemblies from the closed loops to the examination cell and assisting of the disassembly of the fuel within the cell.

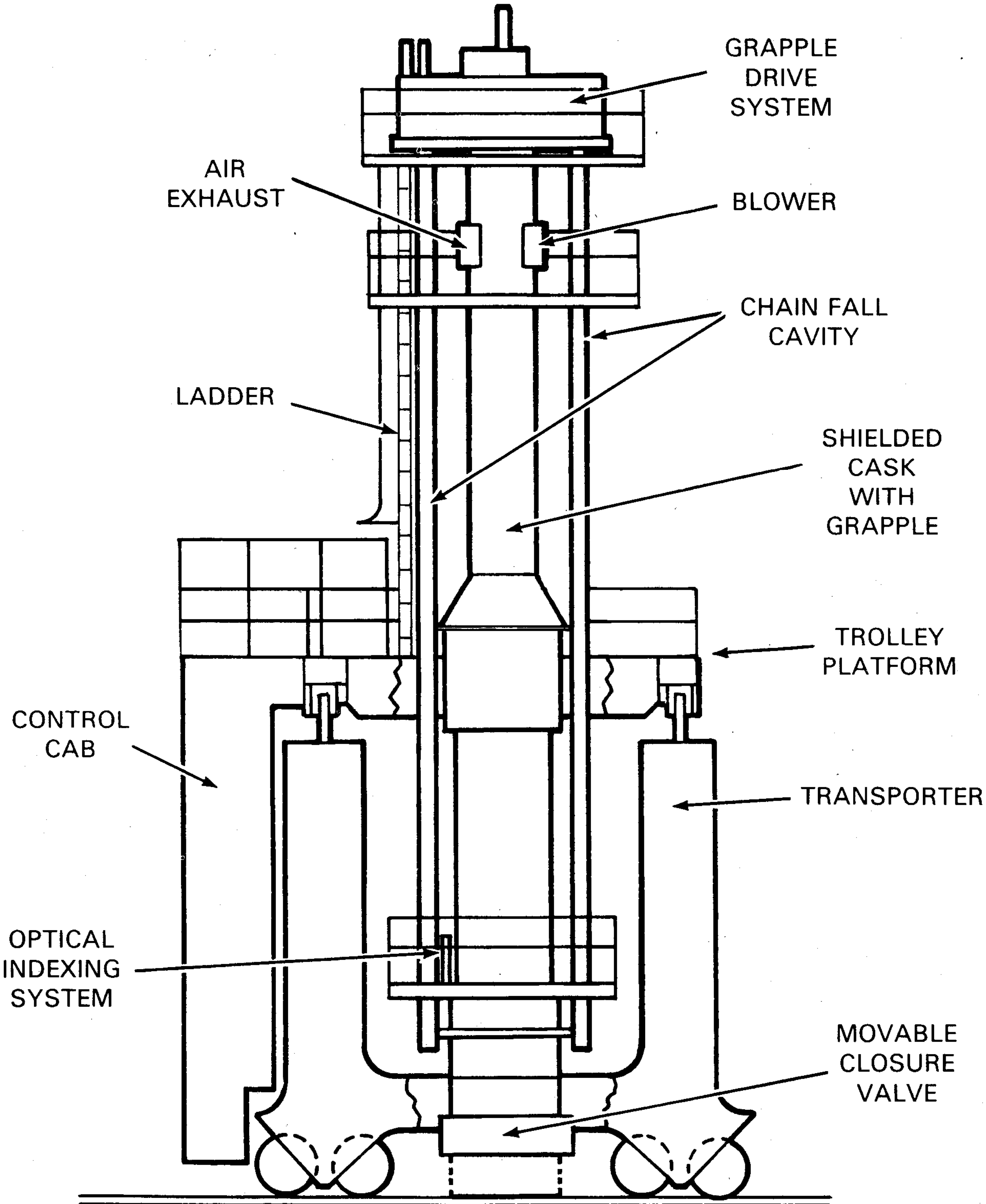

Fig. A.6. Schematic of closed loop ex-vessel machine and transporter.

A.3.2 Bottom-Loading Transfer Cask

The bottom-loading transfer cask (Fig. A.7) transfers core components up to 13.5 ft long between transfer stations inside and outside the FFTF containment building. The transfer cask is a single-barrel, internally cooled machine that travels on rails through the equipment airlock (Fig. A.3) and in-line with the core component conditioning stations (preheating of fresh fuel before fueling) and interim decay storage (cooling of SNF before transfer out of containment). The transfer cask grapple reaches through a floor valve on the operating floor and into each transfer station or cell to install or remove a core component for transfer to another station. The transfer cask is about 10 ft wide by 22 ft long by 19 ft high and is made up of three main parts: the cask, the dolly, and the control system. The total weight of the cask is 72.5 tons.

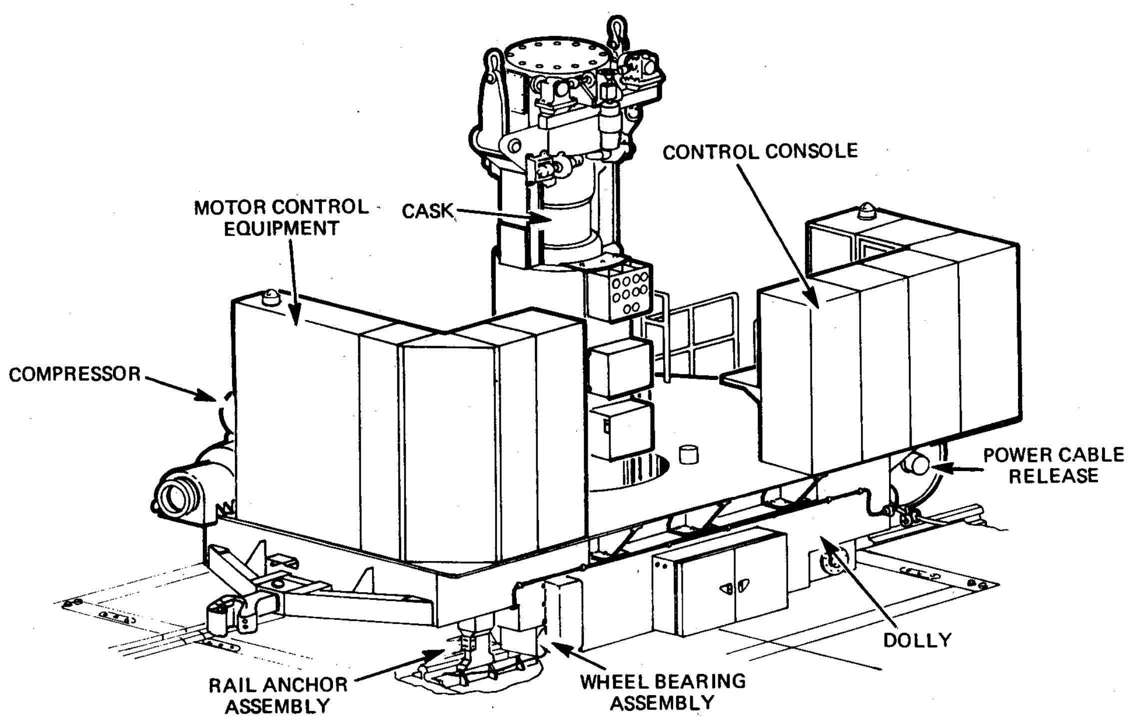

Fig. A.7. Bottom loading transfer cask.

The cask provides the shielding, inert environment, confinement, component grappling, and 1.4-kW(t) cooling capability required to handle sodium-wetted, irradiated core components. Its inside diameter is about 8 in. The dolly provides traverse drive, cask elevation, and seismic-restraint functions. The control system provides the control consoles, motor control equipment, and all interconnecting cables. Control of the transfer cask is semiautomatic; that is, key operations are automatically performed after the operator initiates a command. Interlocks are incorporated to meet safety requirements. Status lights indicate the exact machine conditions during all operations and manual overrides permit the operator to complete key operations in case of a control system failure.

The cask normal cooling mode uses forced argon gas circulated through a closed system. The circulating gas transfers heat from the core component to the cask shielding wall.

A.3.3 Core Component Conditioning Stations

The function of the two core component conditioning stations (Fig. A.8) is to prepare 12-ft-long core components for operation in liquid sodium at elevated temperatures. The components are immersed in an argon atmosphere and heated to 232°C (450ºF). Cooling nitrogen is supplied to the exterior of the separate cells, up through the vaults, to maintain an average cell wall temperature of 38°C (100ºF) throughout the station.

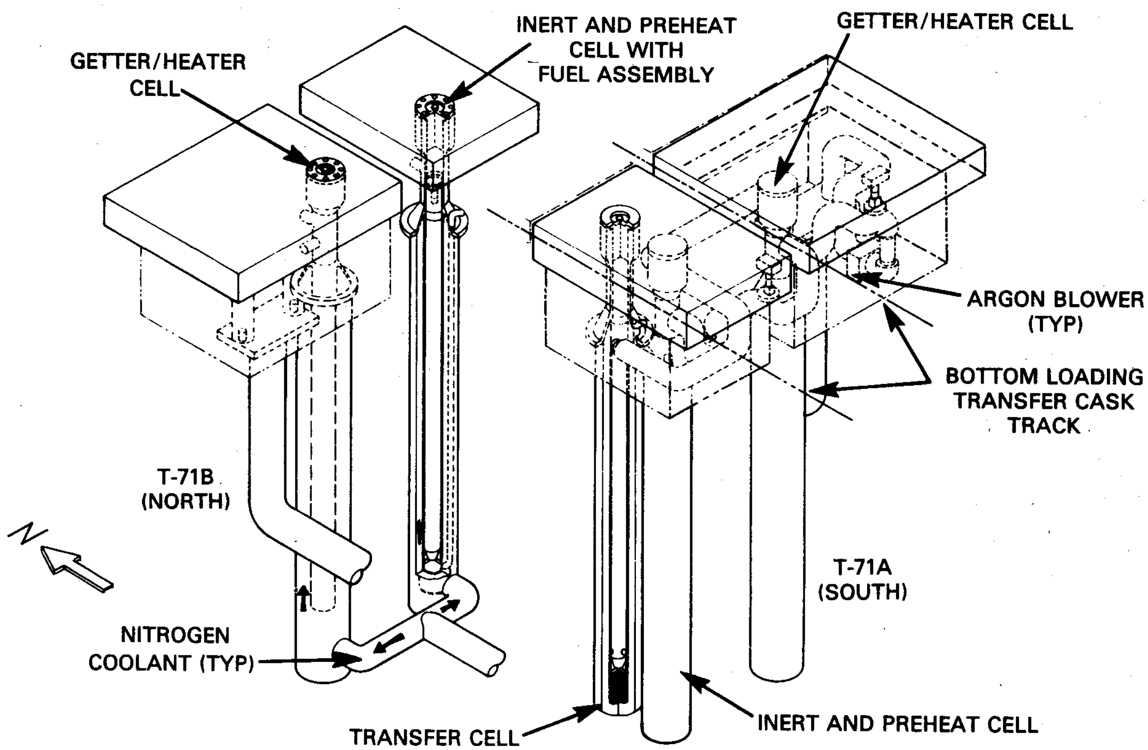

Fig. A.8. Core component conditioning stations.

Each station contains a heater-getter cell for heating the argon and an inerting-preheating cell. One station also contains a transfer cell that permits transfer of up to 13.5-ft long components between the CLEM and the bottom-loading transfer cask.

A.3.4 Test Assembly Conditioning Station

The test assembly conditioning station provides argon flooding and preheating to 232°C (450ºF) for twenty-seven 40-ft-long test assemblies. Space is also provided for storage of radioactive test assemblies, recycled closed-loop in-reactor assemblies, and shield plugs. All storage spaces will accommodate 40-ft assemblies. Design is generally similar to that of the core component conditioning stations. The location of the test assembly conditioning station is shown in Fig. A.1.

A.3.5 Interim Decay Storage

Interim decay storage (Fig. A.9) has two functions.

- *Storage*. It provides a passive, controlled liquid sodium environment for damage-free temporary storage of new components en route to the reactor and for irradiated core components and test assemblies removed from the reactor.

- *Component transfer*. It is the transfer station between the bottom-loading transfer cask, which operates outside the containment area and CLEM, which remains inside the containment area and operates only when the reactor is shut down. Consequently, both the bottom-loading transfer cask rails and CLEM rails traverse the interim decay storage cover.

The interim decay storage facility is in a rectangular steel-lined shielded concrete cell lying entirely below the operating floor, with the top flush with the operating floor. It consists of a rotating storage basket submerged in liquid sodium in a stainless steel tank surrounded by a guard vessel. Storage positions are provided for 112 core components 12 ft long in the upper section of the storage basket. Ten storage tubes are included near the center of the basket to store the 40-ft test assemblies.

The sodium temperature is held between 260 and 316°C (500 and 600ºF). During normal operation, two independent cooling systems are available. The interim-decay-storage sodium-cooling system circulates sodium from the storage tank to an intermediate sodium/sodium-potassium heat exchanger. The secondary sodium-potassium coolant in turn is cooled by a Mobiltherm cooler. If the primary sodium-cooling system is removed from service for any reason, a backup nitrogen-gas-cooling system and an emergency natural-convection air-cooling system may be utilized.

A.3.6 Fuel Storage Facility

The fuel storage facility (Fig. A.10) was designed as a separate project after the FFTF design was completed. The project includes a main 12,200-ft^2 building and a 72-ft by 72-ft extension of the reactor service building. It is designed to store SNF for the first 5 years of FFTF operation. The fuel storage vessel, 21 ft in diameter and 24 ft deep, contains liquid sodium for cooling spent fuel from the FFTF and is surrounded by a guard tank. Two heat exchanger systems dissipate decay heat from the irradiated fuel elements immersed in the sodium for storage.

The 62-ft-high reactor service building extension provides an enclosure for movement of SNF between the FFTF and the fuel storage facility. The building extension also contains a cask load-out station belowground, where SNF elements can be placed in casks for off-site shipment.

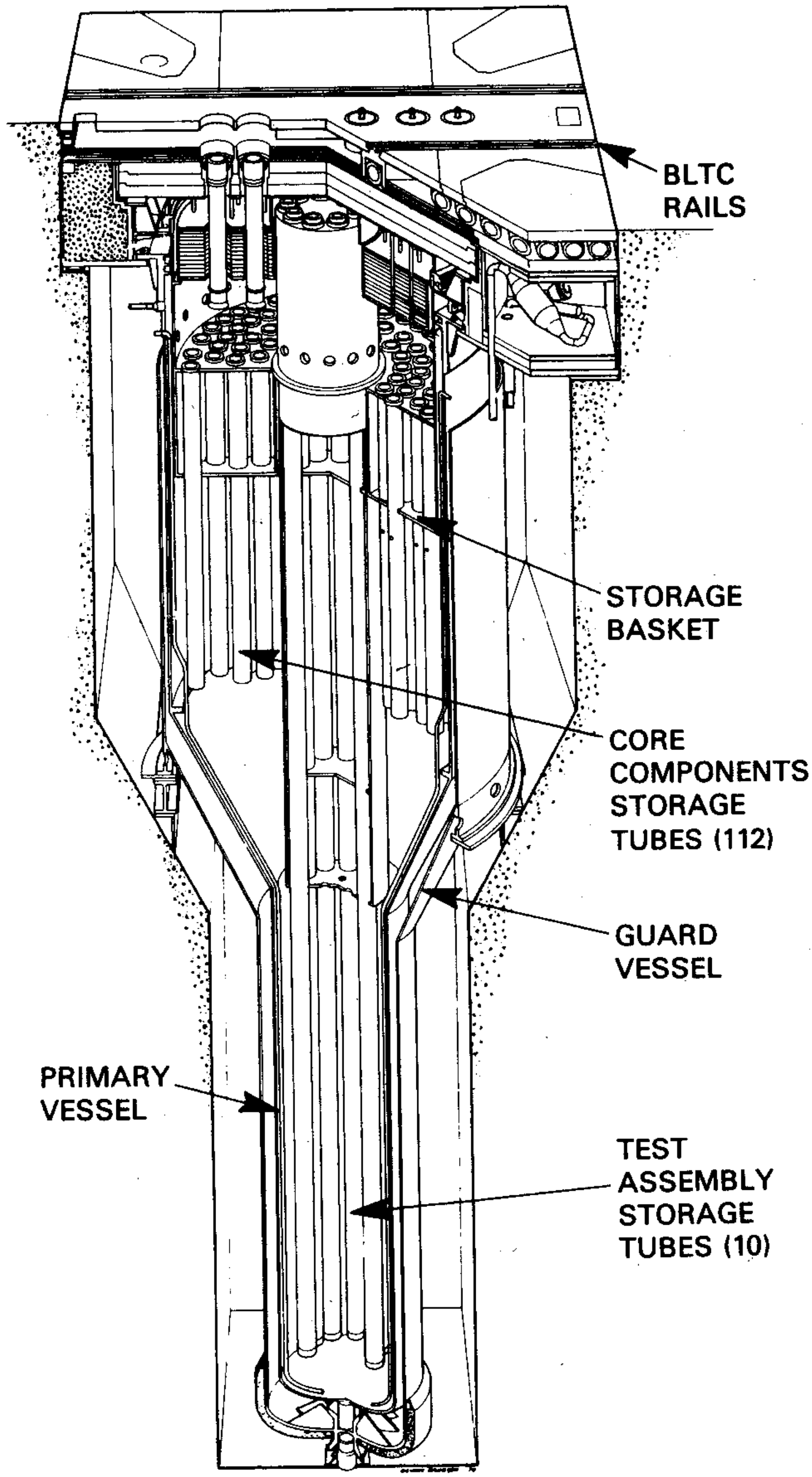

Fig. A.9. Interim decay storage.

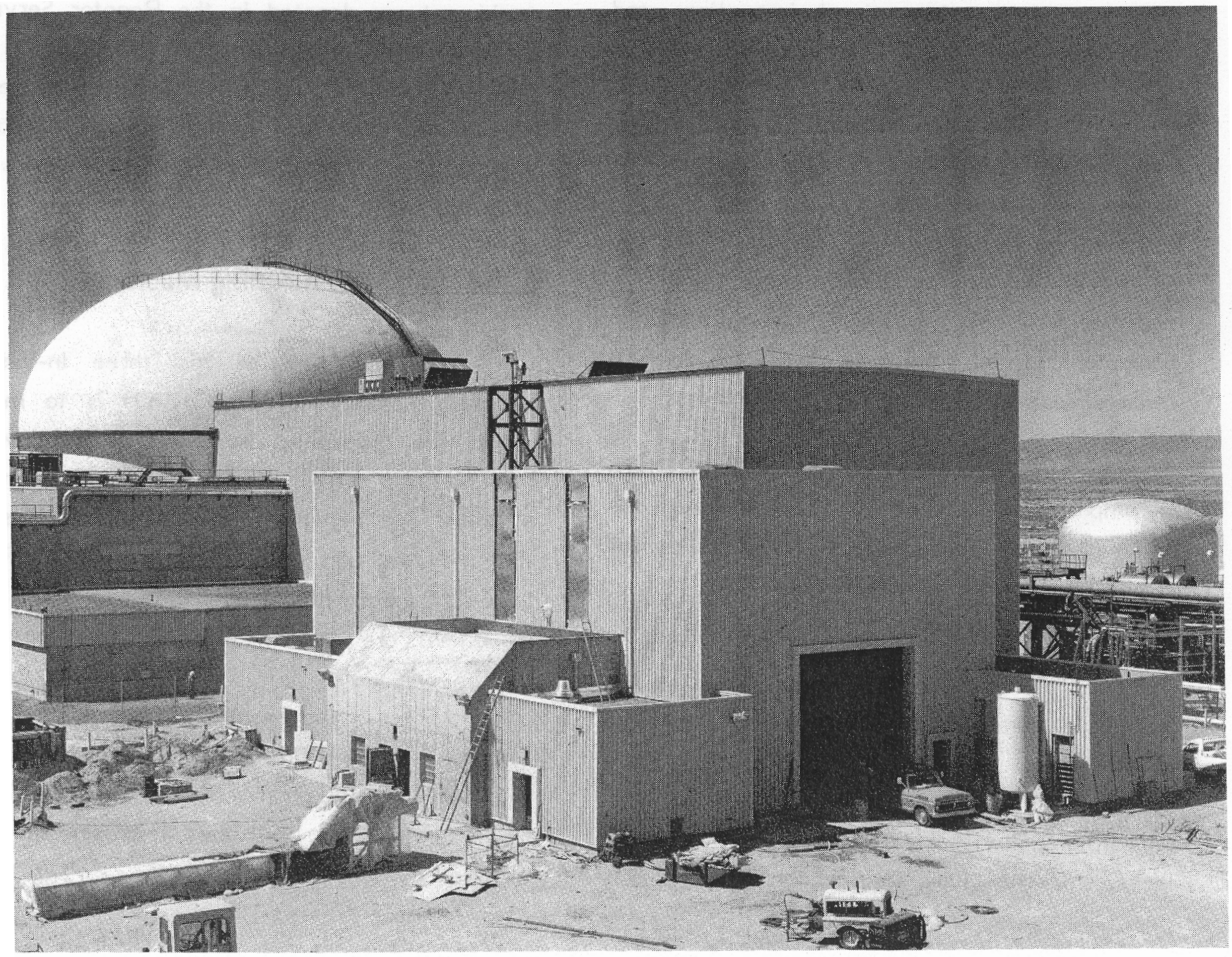

Fig. A.10. Fuel storage facility (under construction).

A.3.7 Interim Examination and Maintenance Cell

The interim examination and maintenance cell is a shielded hot-cell complex that conducts nondestructive examinations of test assemblies and core components under controlled argon-atmosphere conditions. (Its location is shown in Fig. A.1). The hot cell can also be used to perform limited maintenance of plant equipment or to prepare plant equipment for transfer to other maintenance facilities. The cell contains equipment necessary for disassembly, reassembly, and requalification of test assemblies and components. Nondestructive examinations that can be performed include dimensional checks, weighing, gamma scanning, visual inspections, and photography. Sodium removal facilities are also included. Cell ceiling valves and an access plug located over the cell provides for transfer of all core components and maintenance equipment in and out of the cell.

A.3.8 In-Vessel Handling Machines (IVHMs)

The function of the three IVHMs (Fig. A.11) is to move 12-ft core assemblies back and forth between the reactor core and the in-vessel storage modules, and between the storage locations in the modules and the CCPs located beneath the three fuel transfer ports. The transfer ports are the locations where CLEM can transfer a CCP into or out of the reactor vessel. All movements are conducted under sodium.

Each of the three IVHMs services its own 120-degree sector of the reactor core. Most sodium-cooled reactors have a single IVHM or equivalent machine. The FFTF has three machines because various instrumentation systems and special irradiation assemblies restrict access to the top of the reactor core from any one location. The multiple machines are a consequence of the fuel-testing mission of the reactor. Figure A.12 is a plan view showing the locations of the IVHMs and the in-vessel storage modules. Figure A.13 is an elevation view showing one of the IVHMs in its position in the reactor. Each IVHM has associated with it in-vessel storage modules with 26 positions for fuel assemblies.

Each IVHM includes a large plug (known as the reactor refueling plug) installed in the reactor head. Eccentrically mounted on the plug, inside the vessel, is an arm structure. Attached to this structure is a grappling device used to grasp core components. Both the plug and arm structure can be rotated by independent ex-vessel drive systems, and the arm structure can be raised or lowered.

The coordinated movement of the plug and arm structure allows operators to place the grappling device over any core or storage location within the IVHM's sector. The control console employs a computer programmed by a tape containing the sequence of moves for each component being handled. The core component identification technique is based on the use of notches on the outside of the core component-handling socket; these notches are positioned geometrically to represent the serial number in binary count. During reactor operation, the IVHM arm is stored radially adjacent to the core, near the vessel wall.

A.3.9 Shield Plug Handling and Storage

A plug handling fixture removes, transports, and replaces the FFTF shield plugs. This fixture is a portable machine with a winch assembly and grapple enclosed in an airtight container. It is moved by the cranes in the FFTF containment building and is normally stored in the containment building over the shield plug storage station. The plug-handling fixture is nominally 6 ft in diameter, 13 ft high, and weighs 17 tons. The control mode is manual with automatic override shutoff at safety or operational set points. The argon-filled section of the vault contains storage positions for various types of shield plugs. Air cells are provided for general purpose storage.

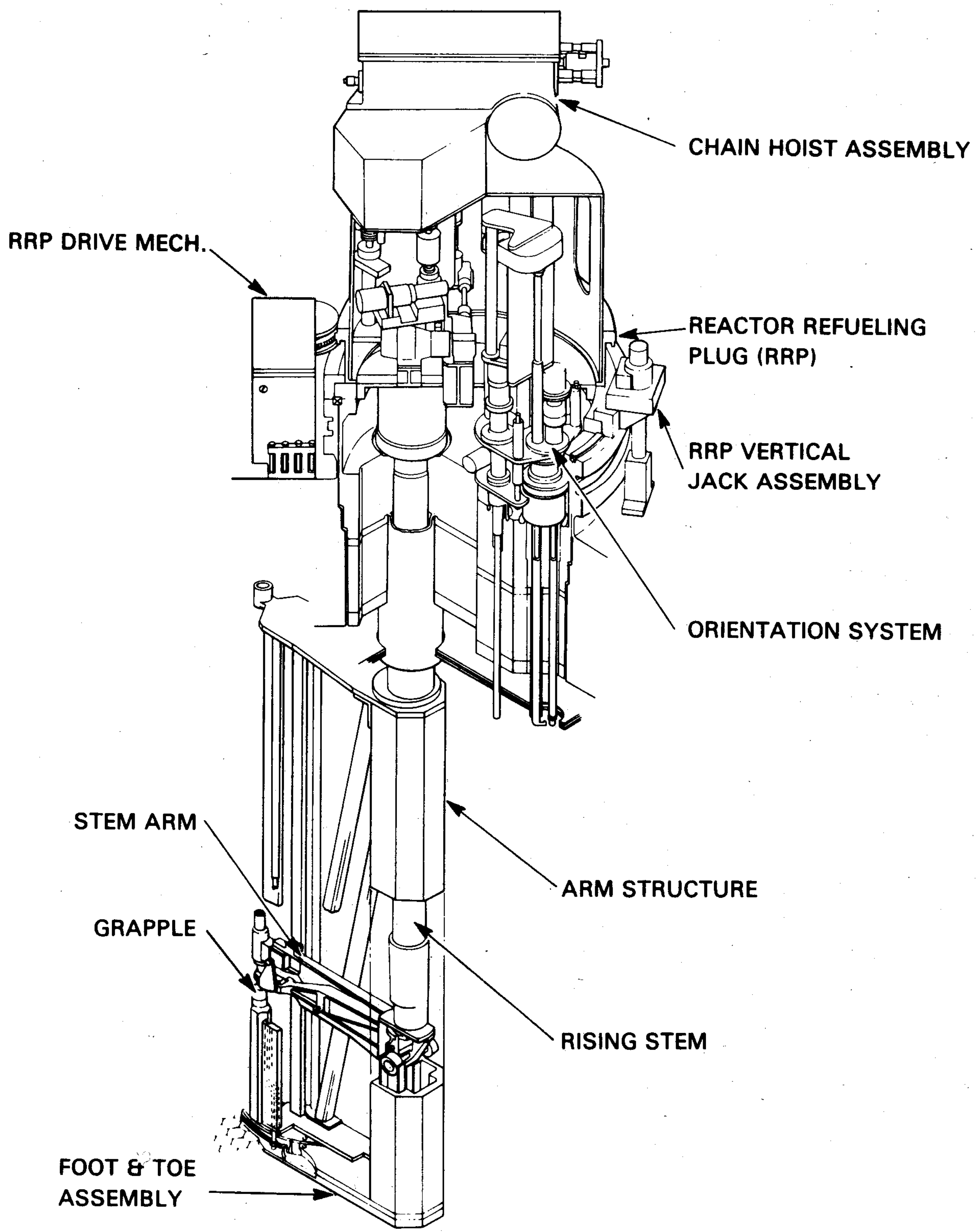

Fig. A.11. In-vessel handling machine.

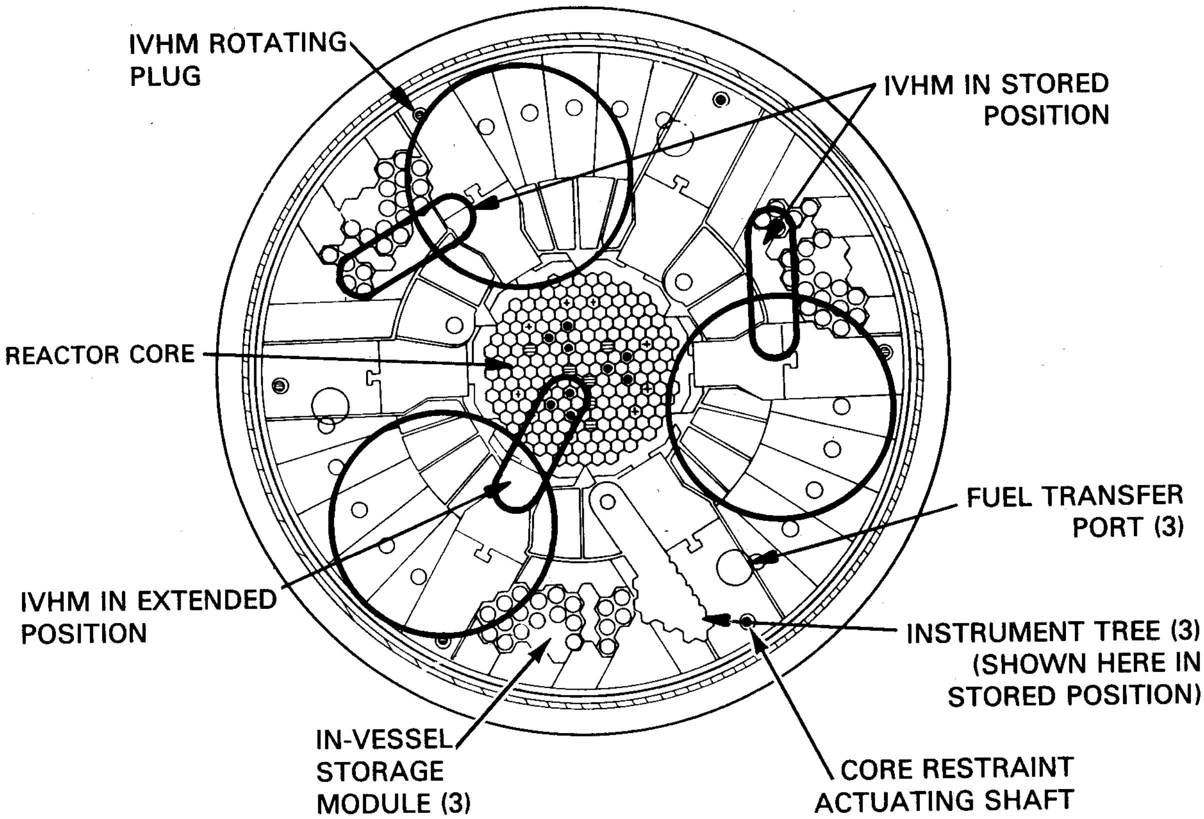

Fig. A.12. Plan view showing locations of in-vessel handling machines.

A.4 SYSTEM THERMAL PROFILE

Core components and test assemblies are exposed to a variety of environmental temperatures as they are routed through the FFTF fuel-handling system. In general, the assemblies going into the reactor are heated, and those being removed are cooled. The transfer between the machine and facility must be a controlled operation; therefore, design points and temperature ranges have been imposed on each item of equipment to provide a nominal thermal profile as shown in Fig. A.14. This figure is provided to allow visualization of temperature conditions that are intended and expected. Some approximations are included in this profile as it covers many diverse conditions that are encountered during normal operation of the system. Even if the incoming assembly is a recycled test assembly, and therefore, has some decay heat, the temperatures are expected to be nominally as shown. The reactor temperature shown in the center of the profile were obtained from the reactor system design description or from reports by the reactor designer but are included here more as reference indicators for the right-hand side of the figure, which covers removal of the core components and test assemblies from the core.

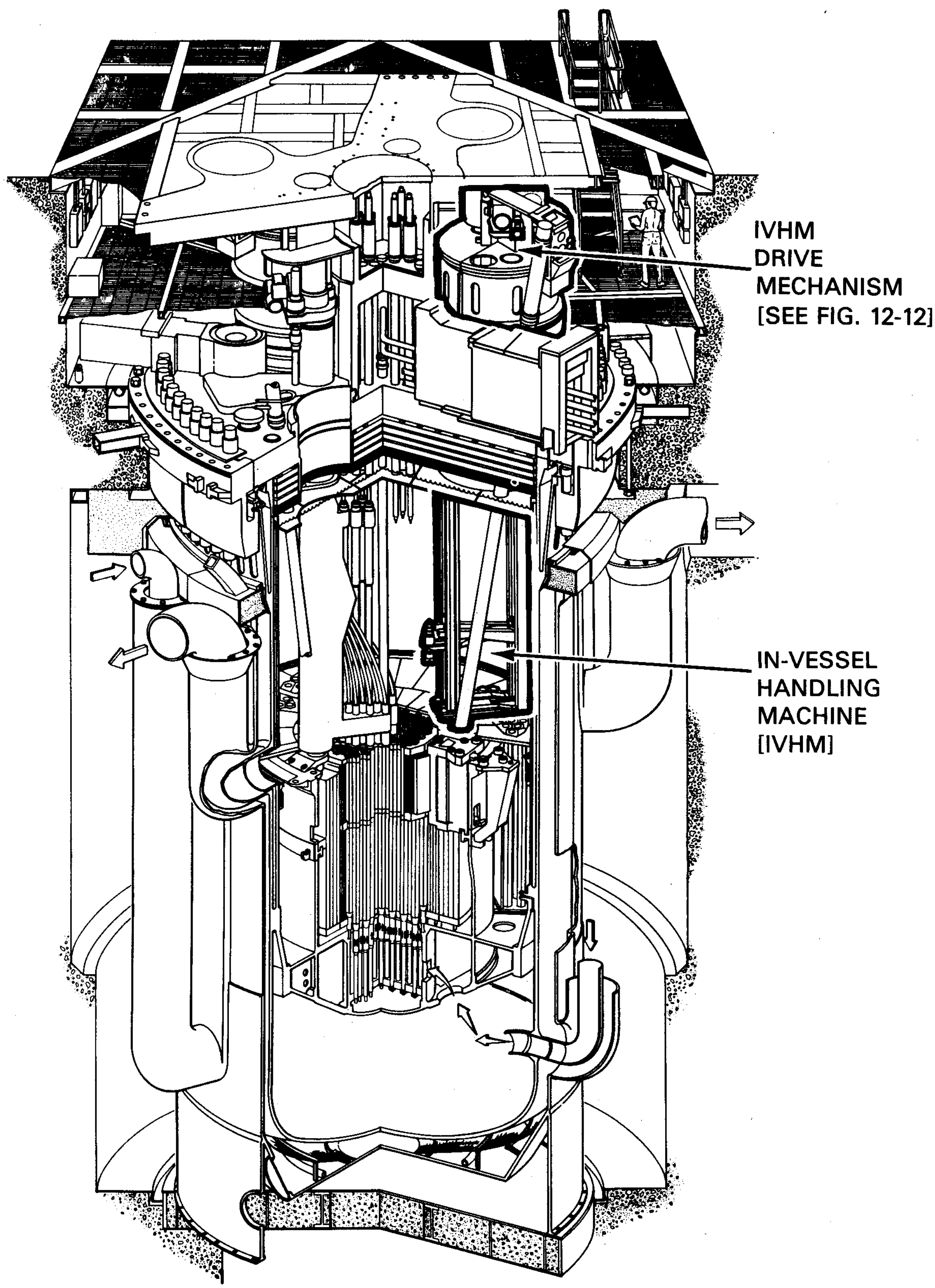

Fig. A.13. Reactor cutaway showing in-vessel handling machines.

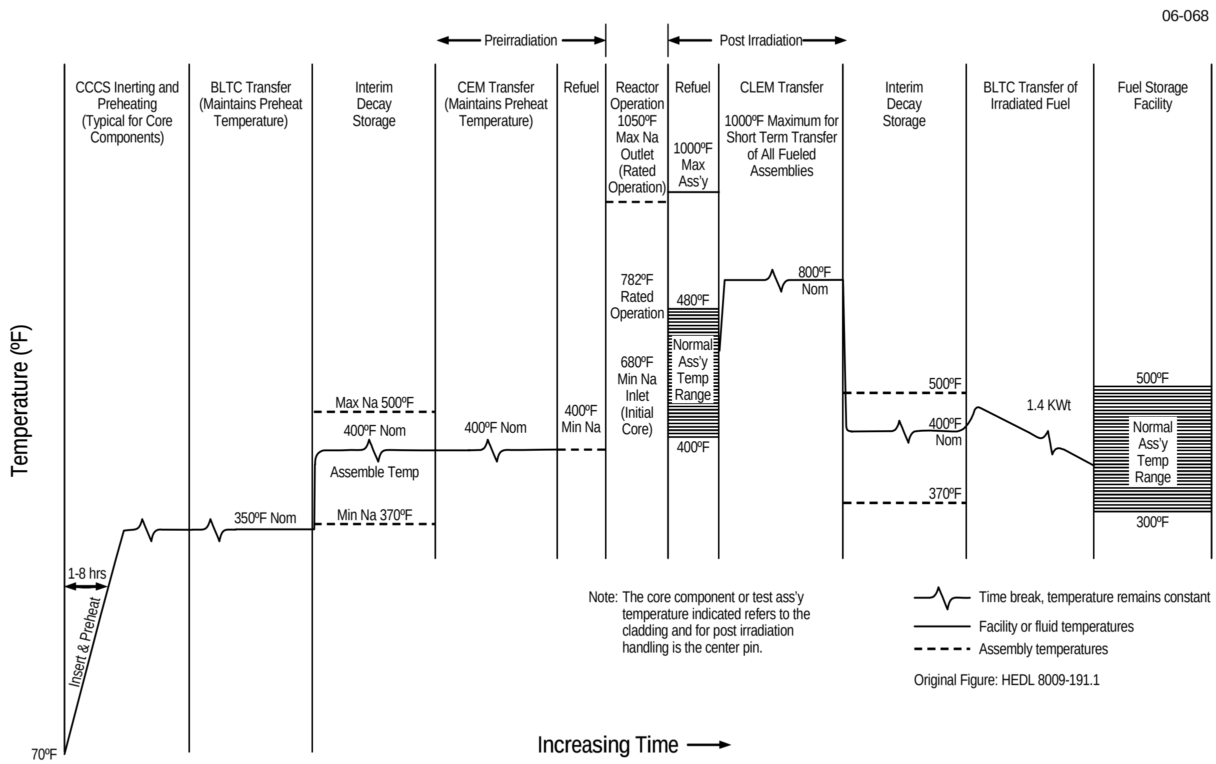

Fig. A.14. Thermal profile for fuel and test assembly handling.

The CLEM transfer of components from the reactor core to interim decay storage is a transient condition, with three cases indicated. As is realistically shown, longer transfer times result in steady-state temperatures that will not compromise the assemblies, and the cases shown represent maximum conditions. Interim decay storage for all but the maximum decay power assemblies (drivers or test assembly) will be nominally at temperatures of 204ºC (400ºF).

The FFTF is a test facility. Thus, the maximum allowable temperatures are significantly higher than the nominal temperatures. Higher temperatures may occur in the transfer and storage systems because of (1) an assembly with particularly high decay heat or (2) an experimental need to avoid cooling of a core component before some type of post-irradiation inspection or test. The post-irradiation conditions for CLEM and the interim examination and maintenance cell are not shown for these cases. The remaining portion of the figure shows the removal of irradiated fuel (drivers or pins) intended for storage. Few limits have been identified for this fuel and cladding. As indicated, the 1.4-kW assemblies will follow the nominal profile, but due to the many variables involved, temperatures could be higher. The temperatures need to be limited only for safety reasons.

References

Cabell, C. P., 1980, A *Summary Description of the Fast Flux Test Facility*, HEDL-400, Hanford Engineering Development Laboratory, December.

FFTF, 1983, *Fast Flux Test Facility System Design Description, No. 41 Part 1*: "Reactor Refueling System," Rev. 11, April 6.

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APPENDIX B: CRBRP FUEL-HANDLING SYSTEM

The design and performance characteristics of the Clinch River Breeder Reactor Plant (CRBRP) fuel-handling system have been reviewed for their relevance to the proposed liquid salt-very high temperature reactor (LS-VHTR). Based on the information detailed in CRBRP documentation (CRBRP 1974a, 1974b, 1983), this summary provides a description of the CRBRP fuel-handling system and its operation.

The CRBRP was designed and licensed as a precommercial prototype power plant, but the plant was cancelled shortly after the start of construction. The refueling systems were designed with strong reliance on Fast Flux Test Facility (FFTF) experience, taking into account the CRBRP mission of power production, in contrast to the FFTF mission of fuel testing. The CRBRP mission required relatively infrequent refueling outages of minimum duration, during which a large fraction of the core would be refueled. In contrast, the FFTF mission required relatively frequent outages, during which both driver fuel and test assemblies were handled. The CRBRP fuel-handling design may be viewed as an evolutionary step beyond the FFTF fuel-handling design toward the development of a fully commercial system.

B.1 CRBRP FUEL HANDLING

The fuel-handling system provides for replacement of the reactor core components, including fuel, blanket, control, reflector, and restraint assemblies. The system consists of the facilities and equipment needed to accomplish the normal scheduled refueling operations, as well as all other functions incident to handling of core components. These latter functions include receiving and unloading of new core components, inspection, temporary storage and conditioning under sodium, transfer of both new and spent core components between storage facilities and the reactor, transfer of core components within the reactor, removal and examination of spent core components, and preparation and loading for shipment of spent core components off-site.

There are three major components of the refueling system, as shown in Fig. B.1.

- *Ex-vessel storage tank (EVST).* This tank is located in the reactor service building and is the primary vessel used to store fresh fuel, spent nuclear fuel (SNF), and other core components. The EVST is a large two-tier, sodium-filled storage tank with a capacity to store two refueling loads of heat-producing (fuel or blanket) assemblies and the heat-producing assemblies from a complete core reactor unloading. Low-heat-producing assemblies may be stored in either the EVST or the fuel-handling cell. The EVST upper tier positions are sufficient to handle a single refueling load.

- *Ex-vessel transfer machine (EVTM).* This rail-mounted transfer cask moves core component pots (CCPs) with fresh fuel and other core components from the EVST to the reactor vessel and moves CCPs with SNF and other core components from the reactor vessel to the EVST. The CCPs are small vessels used to transfer fuel and other components in a temperature-controlled sodium environment. The EVTM is also used for fresh fuel, SNF, and other transfer operations within the reactor service building and remains in the reactor service building during reactor operations.

- *In-vessel transfer machine (IVTM).* This transfer machine is located in the reactor vessel during refueling operations and moves (1) fresh fuel and other core components from CCPs near the outside edge of the reactor vessel to positions in the reactor core and (2) SNF and other used core components from the core to the CCPs.

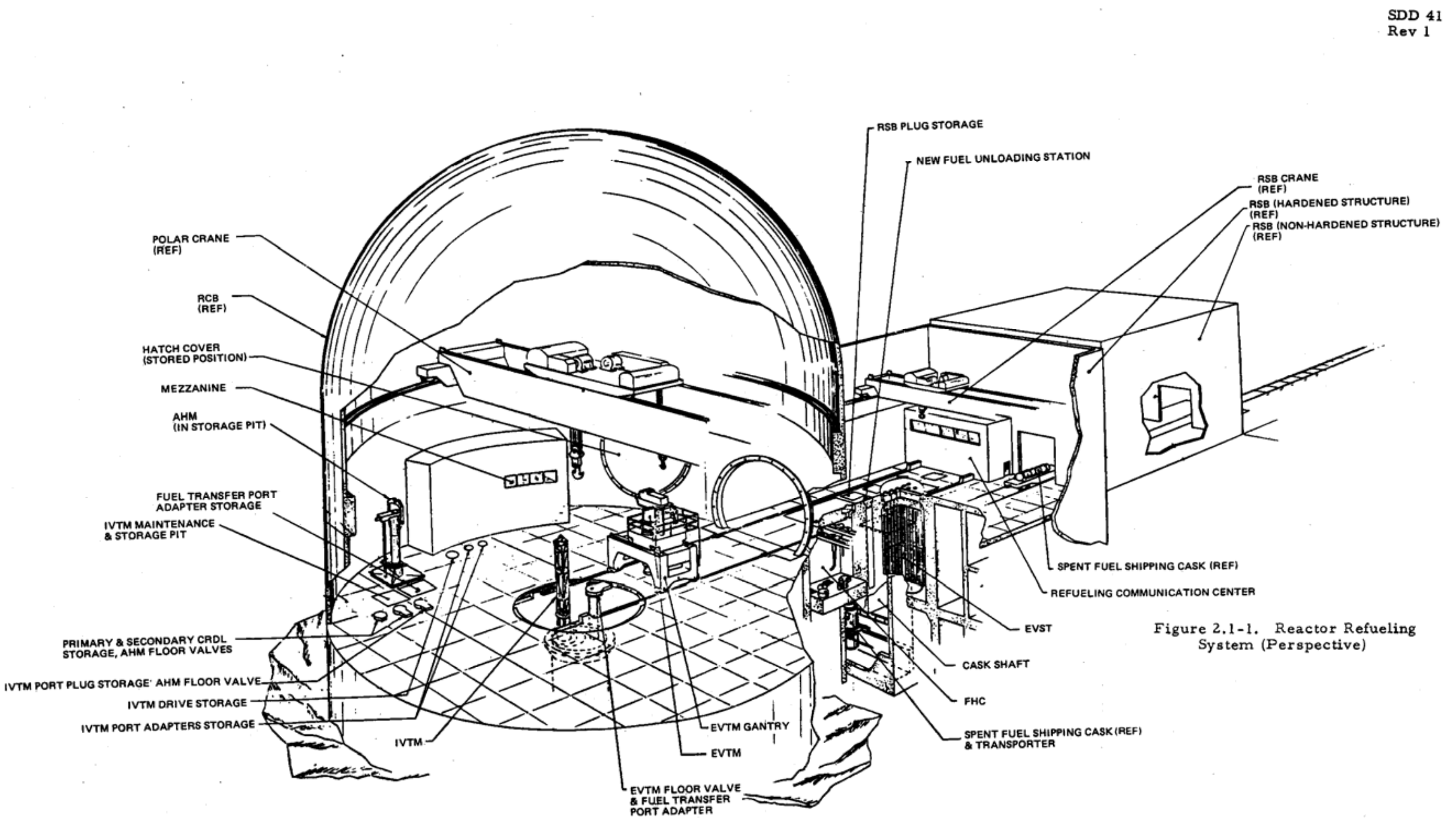

Fig. B.1. CRBRP reactor refueling system.

Each of these major components has multiple support systems, as described in the following sections. The refueling operations can be divided into three steps: preparation for refueling, refueling, and SNF shipment. Only the refueling operations require the shutdown of the reactor.

B.1.1 Preparation for Refueling

New fuel assemblies arrive at the plant by truck in lightly shielded, single-assembly shipping containers. The containers are removed from the truck and placed in a lay-down area. The shock indicators attached to the containers are checked to verify that the containers have not been mishandled or dropped. The reactor service building crane is used to raise a container to a vertical position and the container with fuel assembly is placed into one of two new fuel-unloading stations, which are located between the EVST and fuel-handling cell and are within the coverage of the EVTM.

Following inspection, the atmosphere of the shipping container inside the unloading station is changed from air to argon to be compatible with the EVTM and EVST. The EVTM is mated with the shipping container through an adapter assembly, which is a part of the new fuel-unloading station. The EVTM removes the new core assembly and transfers it to the EVST. The assembly is placed in a preheat tube in the EVST, where it is slowly heated in argon to approximately the same temperature as the sodium in the tank. After preheating, the EVTM is used to transfer the assembly to a sodium-filled CCP in one of the storage positions in the tank. The new core components are loaded into the EVST at 475°F (246°C) until a reactor reload worth of fuel has been accumulated for refueling.

Prior to reactor shutdown for refueling, a number of preparatory operations are conducted to minimize reactor downtime for refueling. All equipment and facilities to be used in the refueling operations are checked for functionality. In addition, actual equipment placement in the reactor service building can begin. This includes placement of floor valves over two EVST fuel transfer ports, removal of the port plugs by the EVTM, and transfer of the plugs to the reactor-service-building plug storage facility.

B.1.2 Refueling

After the reactor is shut down, the equipment hatch between the reactor service building and the reactor containment building is opened. The sodium in the reactor vessel is cooled down to refueling temperature, the control rod drive lines are disconnected from the absorber assemblies and raised, and the upper internals are raised, permitting rotation of the plugs in the reactor vessel head (Fig. B.2). At the same time, the reactor cover gas is purged and purified to reduce radioactivity levels in the gas to a very low level. This completes the preparations for refueling and permits reactor refueling operations to begin.

Reactor refueling operations during reactor shutdown begin with installation of adapters and floor valves over the reactor fuel transfer port and the IVTM port in the reactor vessel head, using the reactor-containment-building polar crane. The various port plugs are then removed by the auxiliary handling machine and the EVTM and are placed in plug storage locations within the reactor containment building and reactor service building. The IVTM in-vessel section is then installed using the auxiliary handling machine. At this point, the floor valve and adapter on the IVTM port are removed and the IVTM drive section is installed; all these operations are performed by the polar crane. The IVTM then undergoes a simple checkout, to ensure that all electrical and gas services have been connected properly, followed by a check of the reactor lid rotating plugs, to assure that core positions can be properly indexed. This completes the preparations of the reactor refueling systems for the process of refueling.

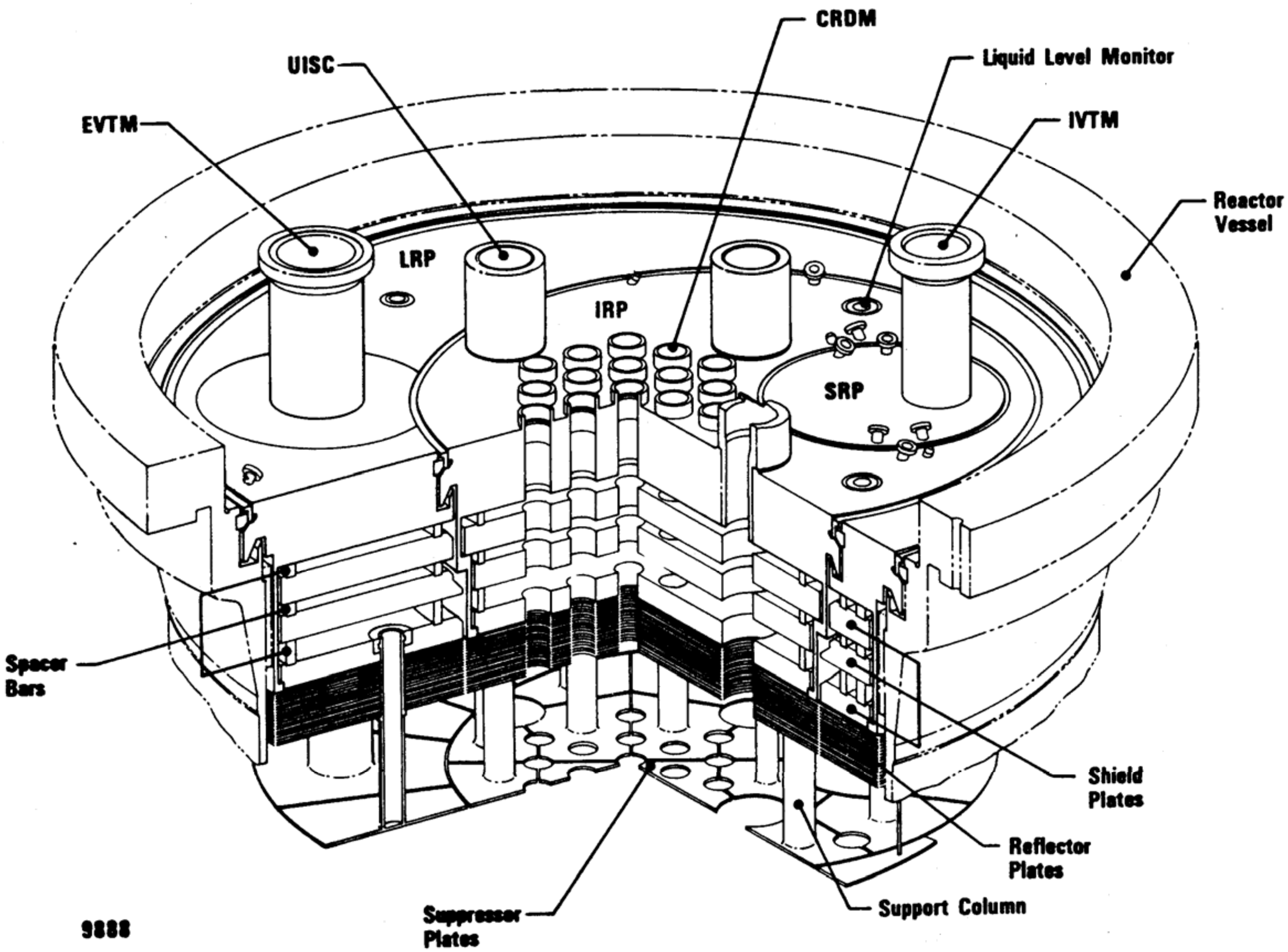

Fig. B.2. CRBRP reactor-vessel-head rotating plugs.

In the process of refueling, the EVTM first moves an empty sodium-filled CCP from the EVST into a transfer position on the periphery of the reactor core. The EVTM then returns to the reactor service building and removes a new fuel assembly in a sodium-filled CCP from the EVST while the IVTM in the reactor vessel removes an SNF assembly from the reactor core and deposits it into the empty CCP. The EVTM then places a CCP with a new fuel assembly in a second transfer position inside the reactor vessel and removes the sodium-filled CCP with the SNF assembly. The SNF assembly is transferred to the EVST. The cyclic process continues until refueling is completed.

The actual transfer of assemblies in the reactor vessel by the IVTM is a multistep process. The IVTM is a vertical push-pull machine mounted on the small rotating plug which forms part of the triple rotating plug assembly in the head of the reactor vessel (Fig. B.2). By grappling a fuel assembly, raising it above the core with the IVTM, and rotating the plugs, SNF is transferred from a core position to a transfer position within the reactor vessel. The SNF is deposited into a CCP, which is then picked up and removed by the EVTM. The IVTM can be indexed over any core or transfer position in the reactor. When the EVTM is at the reactor, the IVTM and rotating plugs must remain stationary.

After refueling is completed, operations to terminate refueling begin. The process used to terminate refueling is essentially the reverse of that used to prepare for refueling. The upper internals are lowered, control rod drive lines are lowered and reconnected, and the reactor instrumentation is checked out. The equipment hatch is then closed and leak checked, and the reactor is made critical and returned to full power.

B.1.3 Spent Fuel Storage and Shipping

After spent fuel has decayed for about 100 days in the EVST, it may be loaded into the SNF shipping cask. Control, radial shield, and some low-power blanket assemblies can be shipped off-site before the 100-day cooling period, but fuel and high-power blanket assemblies are held until they decay to within the spent fuel shipping cask heat rejection capability.

B.2 SYSTEM DESIGN REQUIREMENTS AND FEATURES

To minimize power costs, refueling downtime must be minimized. Twenty days has been allotted for an average reactor refueling. This begins with reduction of reactor power from 100% to the power level at which the reactor is shut down. Two strategies are used to minimize refueling downtime.

- *Maximize off-line operations*. The majority of the refueling equipment and facilities are located in the reactor service building, which is adjacent to the reactor containment building. This maximizes the number of operations and maintenance that can be performed while the reactor is still operating. Important fuel-handling facilities and equipment located in the reactor service building include the EVST, the fuel-handling cell, new and spent fuel unloading and loading facilities, miscellaneous storage facilities, and a communications center from which refueling and other fuel-handling operations are coordinated. Equipment and facilities located in the reactor containment building are limited to those involved with fuel handling in the reactor; they include the IVTM, the auxiliary handling machine, and miscellaneous storage facilities. Unlike FFTF (Appendix A), the IVTM does not remain in the reactor during operations. This allows checkout and maintenance of the IVTM during reactor operations.

- *Efficient refueling*. The design of the refueling system minimizes refueling time by minimizing the number of operations that must be done during the refueling outage. In the sodium-cooled fast reactor, where the decay heat levels per SNF assembly are very high, a key part of the strategy is movement of the SNF in CCPs. These small sodium-filled transfer vessels keep the SNF under sodium, provide the necessary SNF cooling, and minimize SNF handling. Any SNF inspections or sodium removal operations are then conducted after the reactor is back online.

B.3 SYSTEM COMPONENTS

B.3.1 In-Vessel Transfer Machine

The IVTM transfers fuel and core components to and from the reactor core to positions at the edge of the reactor vessel where the EVTM can reach them. The IVTM has following operational capabilities:

1. Grapple and release core assemblies
2. Raise and lower core assemblies
3. Provide hold-down of adjacent core assemblies
4. Uniquely identify core assemblies
5. Orient and center core assemblies for insertion in the core

The IVTM consists of two principal subassemblies (Fig. B.3). The in-vessel section contains the grapple, centering device, identification mechanism, the hold-down mechanism, and the seals for reactor cover gas containment. The ex-vessel section contains the drive equipment, which powers the in-vessel section mechanisms. The IVTM is a straight push-pull machine; horizontal translation is accomplished by rotation of the plugs in the reactor head.

The lower portion of the in-vessel section is exposed to reactor sodium and the cover gas. It is installed and removed using the auxiliary handling machine, which provides a shielded inert-gas environment. The drive section is exposed only to the reactor-containment-building air environment and can be installed using the polar crane. All parts requiring routine service or inspection are located on the drive section and on the in-vessel section above the reactor head to provide for easy accessibility. The dynamic elastomeric seals, which are located in the upper end of the in-vessel section, are continuously monitored, but any necessary replacement of these seals will be done at the facility used to store the IVTM in-vessel section.

The major performance characteristics of the IVTM are summarized in Table B.1.

B.3.2 Auxiliary Handling Machine

The auxiliary handling machine (Fig. B.4) is a single-barrel handling machine that provides an inert atmosphere and radiation shielding for sodium-wetted radioactive components being transferred from the reactor and to storage facilities within the reactor containment building. These operations primarily involve preparation for (and termination of) refueling and include (1) removing the IVTM port plug and installing the IVTM in-vessel section and (2) returning these items after refueling. The auxiliary handling machine is also used to support maintenance functions such as removal and installation of control rod drive lines and core inlet modules. The auxiliary handling machine is similar in concept to the EVTM, but it has been simplified due to the less-demanding requirements of the components it handles. The major differences are (1) no systems are needed to manage radioactive decay heat, (2) less radiation shielding is required, (3) fewer requirements are associated with the management of sodium drainage from components, and (4) less stringent requirements exist for speed of operation. The auxiliary handling machine is used only within the reactor containment building and is transported from station to station by the polar crane.

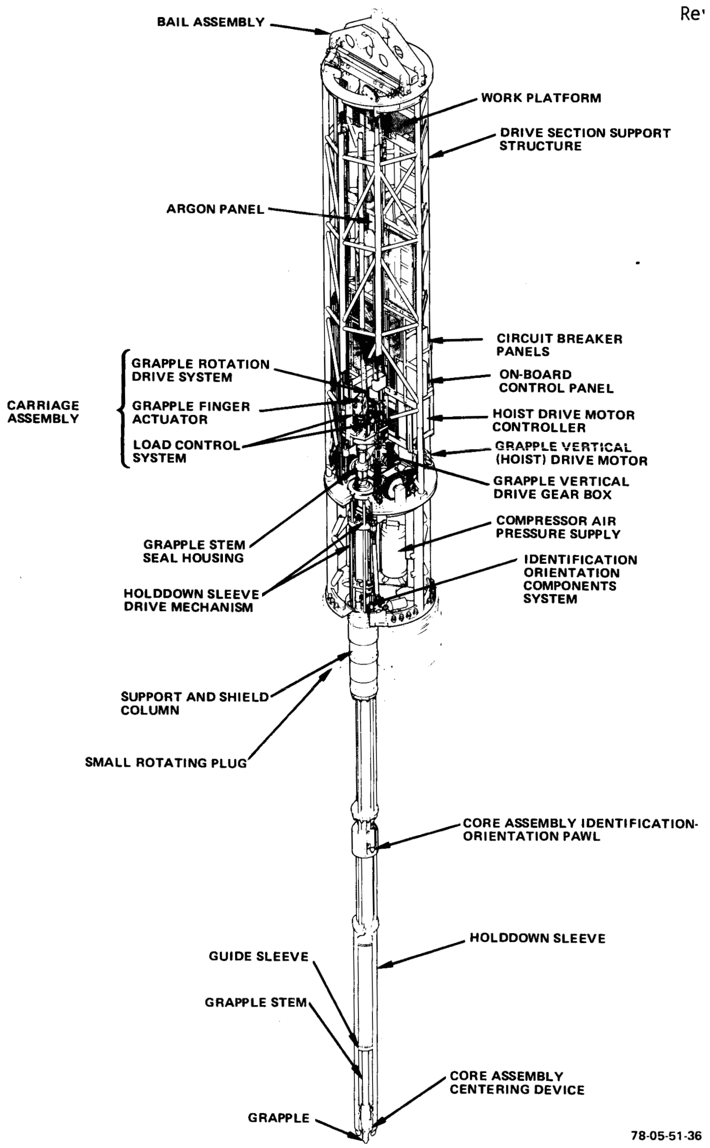

Fig. B.3. In-vessel transfer machine.

Table B.1. Characteristics of the IVTM[a]

Parameter	Value
Grapple vertical stroke	~18 ft
Core hold-down stroke	~3-1/2 ft
Grapple vertical speed (approximate)	~1 to ~15 ft/mn
Grapple vertical load	
Normal operational limit	1000-lb pull 1000-lb push
Maximum operational limit	4300-lb pull 3000-lb push
Design capability	5000-lb pull 5000-lb push

[a]Design concept; straight push pull, rising stem, with triple rotating plugs

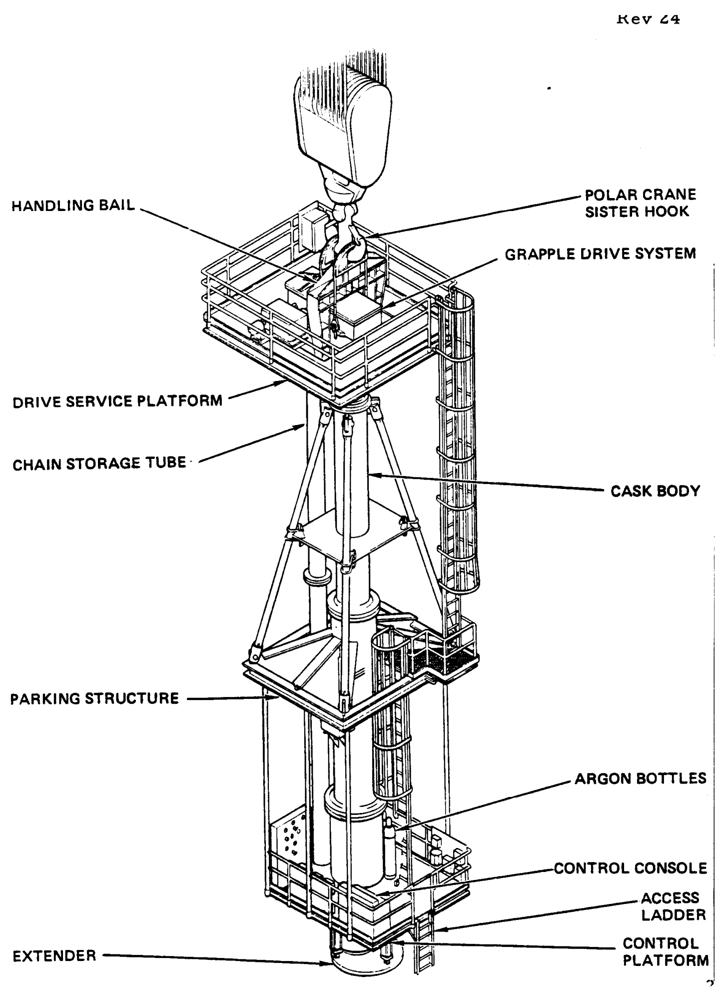

Fig. B.4. Auxiliary handling machine.

B.3.3 Floor Valves

The function of the floor valves (Fig. B.5) is to seal a reactor port or other facility access ports during various phases of the refueling operation when the port plug is removed. The floor valves prevent the release of radioactivity-contaminated gas and shield the building operating floor environment from radiation. Floor valves are portable items of equipment. Two basic sizes are used─one for the transfer of core assemblies (used with the EVTM) and another with a larger opening (used with the auxiliary handling machine) for the transfer of the in-vessel section of the IVTM and other components. Both types have the same basic design, except for the interior opening size and differences in the thickness of the shielding surrounding the central cavity. To simplify nomenclature, the portable floor valves are identified as EVTM floor valves and auxiliary-handling-machine floor valves, depending on the principal mating machine. The auxiliary-handling-machine floor valves are normally used at three locations in the reactor containment building during refueling—the IVTM port on the reactor head, the IVTM port plug storage pit, and the IVTM storage and maintenance facility. These valves are also stored in the reactor containment building. The floor valve design is similar to that developed for the FFTF program.

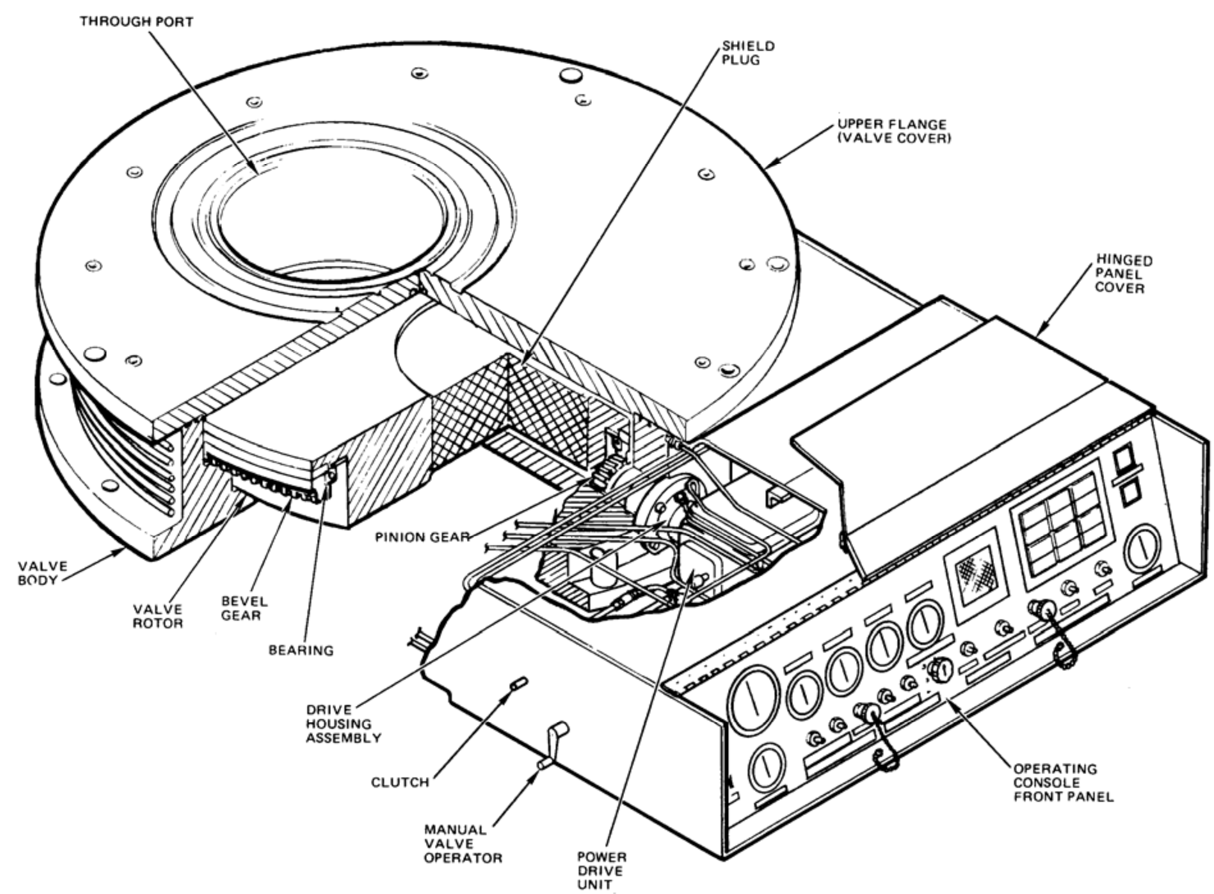

Fig. B.5. Auxiliary-handling-machine floor valve.

B.3.4 Port Adapters

The port adapter provides an isolated shielded area connecting a port in a facility located below operating floor level and a floor valve placed at floor level, through which contaminated and/or radioactive components may be safely transferred. Adapters are sized for specific applications and may be generally described as hollow, open-ended, thick-walled stainless steel cylinders equipped with flanges and seals at each end. One end mates with and seals to a facility port, while the other end mates with and seals to a floor valve. Figure B.6 shows the IVTM port adapter installed on the reactor small rotating plug.

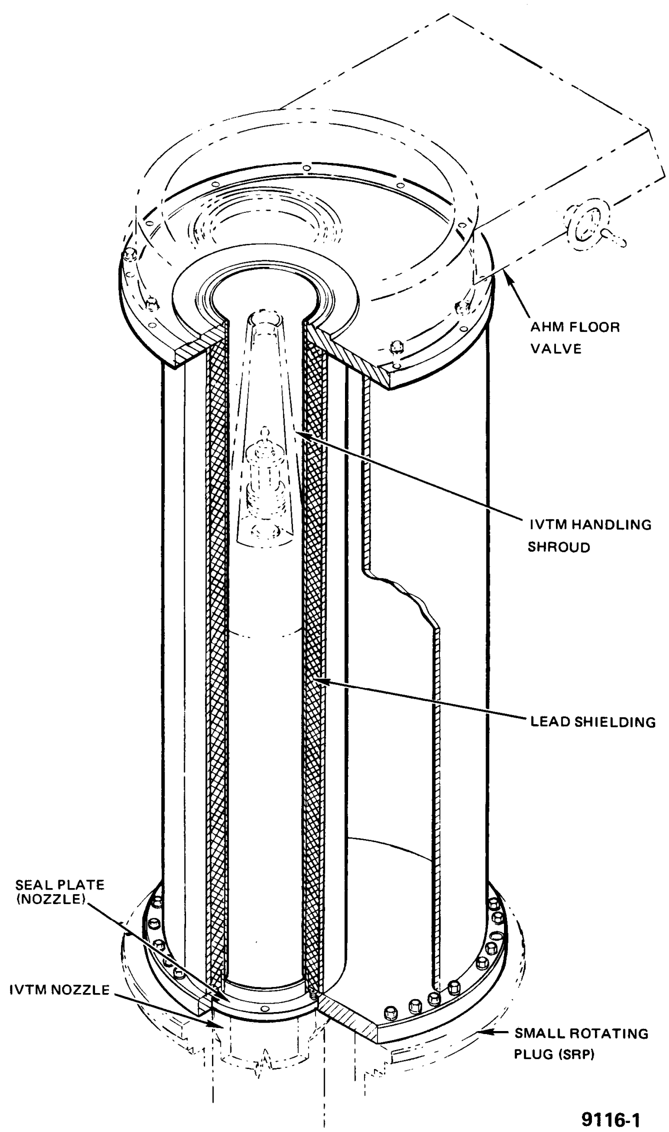

Fig. B.6. IVTM port adapter.

B.3.5 Ex-Vessel Transfer Machine

The primary function of the EVTM is to transfer both new and irradiated core assemblies between various stations. The following operational capabilities enable the EVTM to carry out this function:

1. Grapple and release core assemblies, core component pots, and port plugs
2. Raise and lower core assemblies, core component pots, and port plugs
3. Provide containment of radioactive cover gas
4. Maintain an argon environment
5. Maintain preheat temperature for new core assemblies
6. Provide cooling for SNF assemblies
7. Provide radiation shielding

The EVTM, shown in Fig. B.7, is a shielded, inerted, single-barrel machine. The major performance characteristics of the EVTM are summarized in Table B.2. The machine has provisions for maintaining the temperature of the new core assemblies and for cooling SNF assemblies. The EVTM is mounted on a trolley which is positioned on rails on top of a gantry (Fig. B.8). The gantry moves on crane rails between the reactor service building and the reactor containment building. The trolley rails are perpendicular to the gantry rails, allowing complete indexing of the EVTM.

Table B.2. EVTM performance characteristics

Design concept	Cold wall, single barrel
Method of transport	Gantry-Trolley
Heating	Electrical
Cooling	Cold wall
Design capability	20 kW
Grapple drive and actuation	Chains
Grapple vertical force capability	~3500-lb pull ~500-lb push
Grapple speed	~24 ft/min (maximum)
Components handled	CCP, core assemblies, port plugs

Irradiated SNF assemblies are transferred only in sodium-filled CCPs. Cooling of the EVTM is accomplished by a cold-wall concept consisting of an axial-finned tube, which also provides containment for the argon cover gas, and a forced-convection air circulation system. The cooling concept is illustrated in Fig. B.9. Heat from the 3-ft-high fueled region of the SNF assembly is distributed over the ~15-ft length of the CCP by natural convection of the sodium in the pot. Heat is transferred to the cold wall primarily by thermal radiation and secondarily by conduction across a stagnant argon-filled gap. The cold wall is cooled by forced convection of ambient air circulated past the axial fins. This cooling system has a blower with the capacity to circulate sufficient air to maintain the fuel cladding temperature below the normal temperature limit. In case of blower failure or complete loss of all power, natural-convection airflow is sufficient to maintain the fuel cladding temperatures below its fuel failure limit.

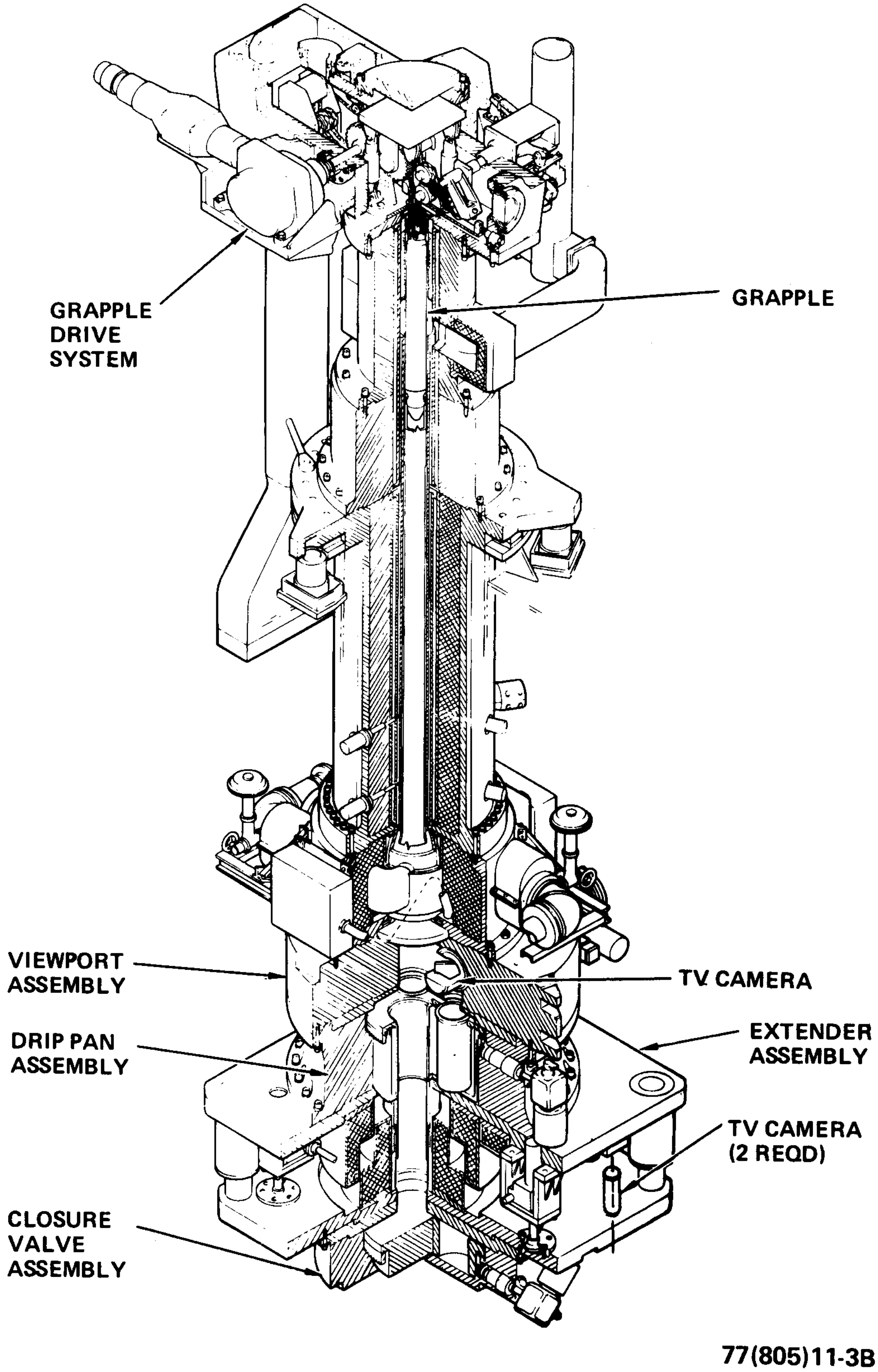

Fig. B.7. Ex-vessel transfer machine.

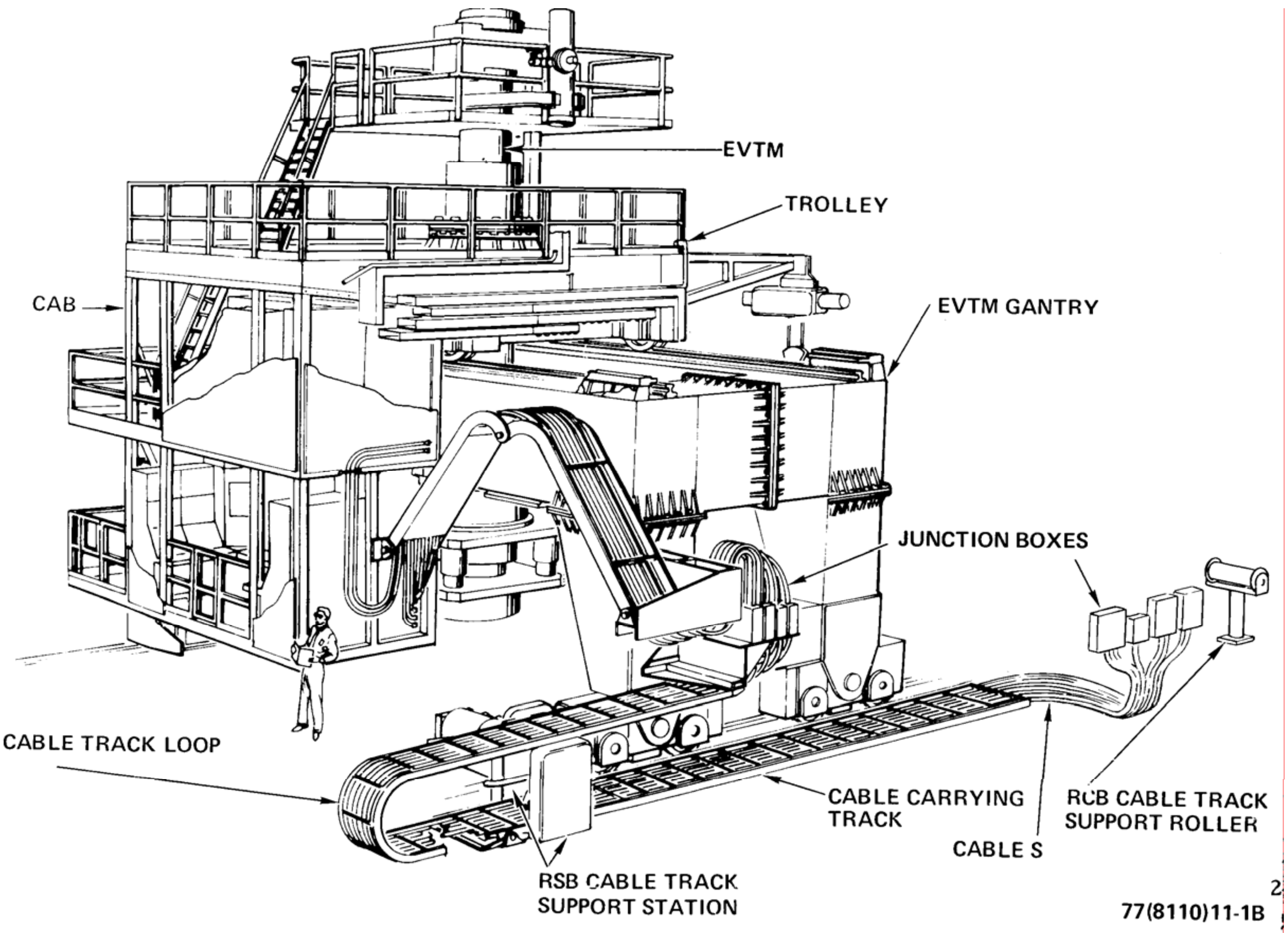

Fig. B.8. EVTM gantry-trolley.

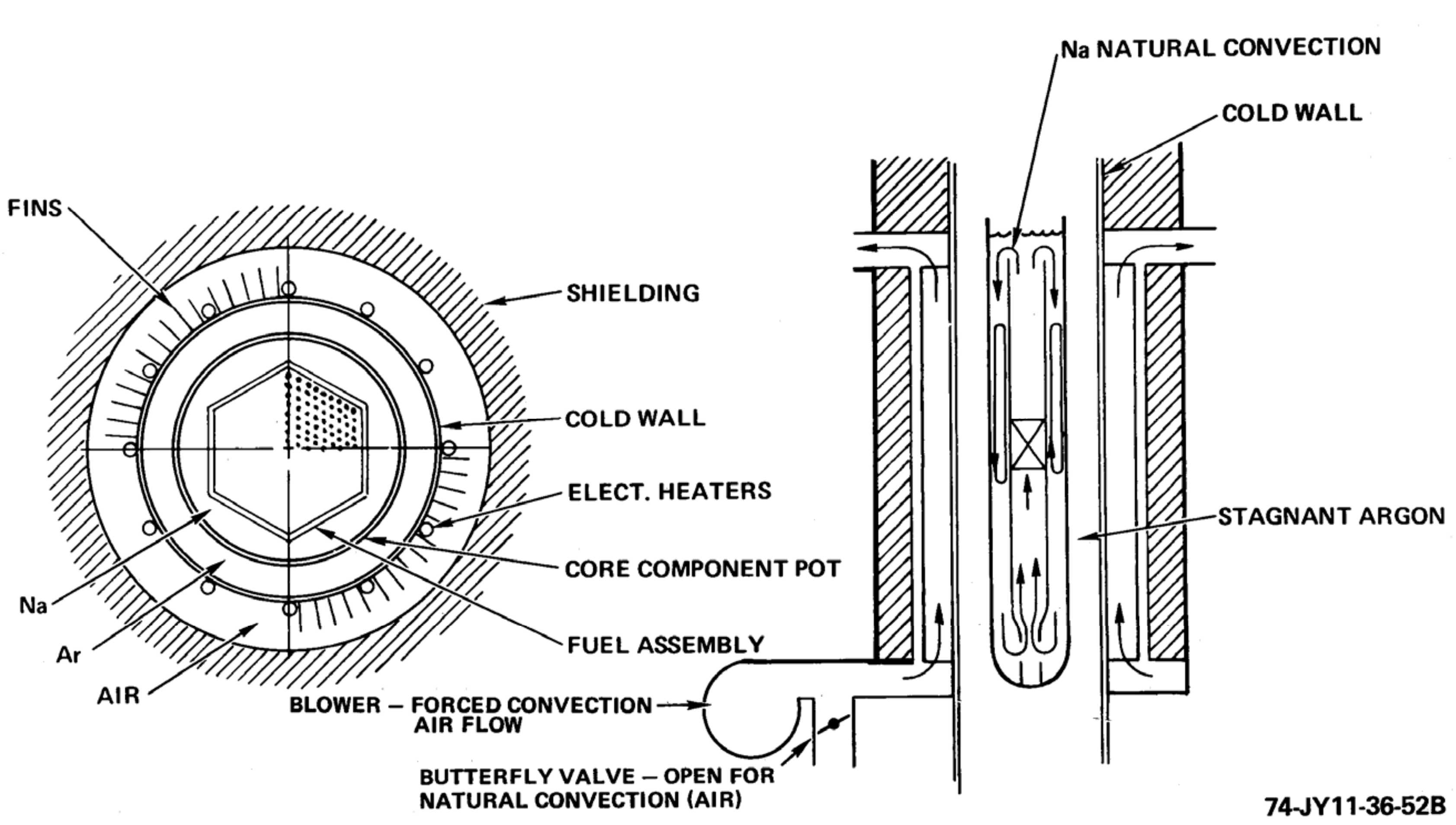

Fig. B.9. EVTM cold-wall cooling concept.

B.3.6 Core Component Pot

The function of the CCP is to provide a temperature-controlled sodium environment for core assemblies being transferred to and from the reactor vessel and fuel-handling facilities. The CCP (Fig. B.10) is a tubular steel container, closed at the bottom, with a handling socket at the upper end which can be grappled by the EVTM or the fuel-handling cell grapple. A siphon assembly near the top of the CCP is designed to remove a predetermined volume of sodium at the time the pot is raised above the facility sodium level, thereby allowing for subsequent sodium thermal expansion due to the decay heat of the SNF. This feature prevents overflow and greatly reduces sodium drippage and splash, as well as the consequent operational and contamination problems. The CCP is used to receive core assemblies during in-vessel transfer, to transport them in the EVTM, and to store them in the EVST and fuel handling cell. A CCP is needed for every storage position in the EVST.

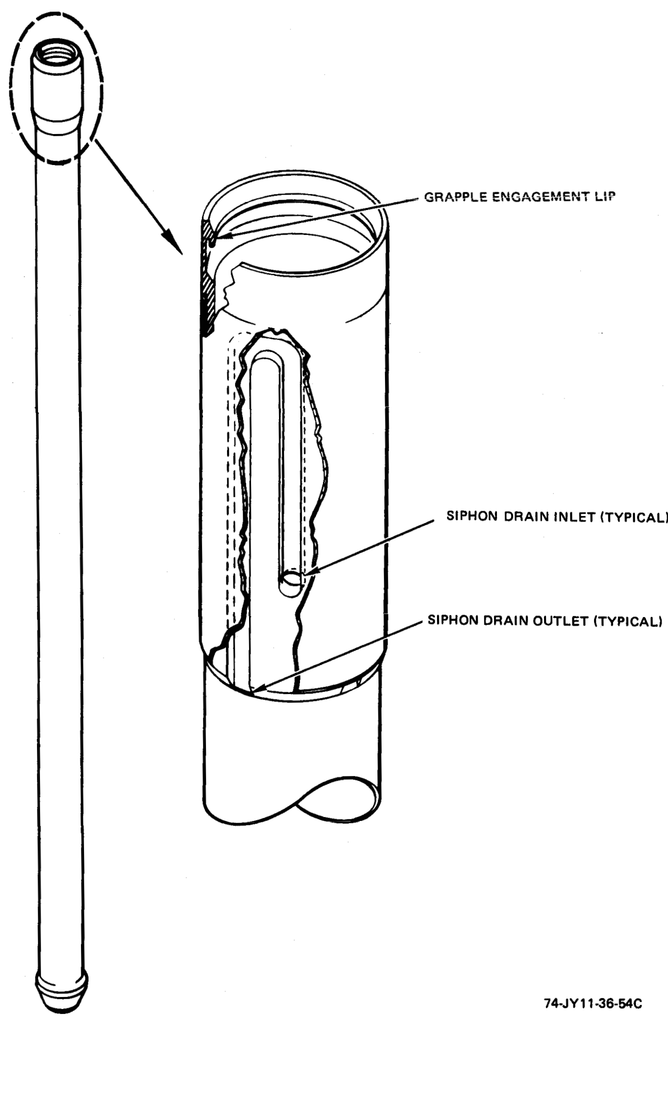

Fig. B.10. Core component pot.

B.3.7 Reactor Fuel Transfer Port Adaptor

The reactor fuel transfer port adapter (Fig. B.11) consists of two assemblies: an upper adapter and a lower adapter. The lower steel adapter is semipermanently bolted to the reactor head at the reactor fuel transfer port. The removable steel and lead-shielded upper adapter is used during refueling and extends from the lower adapter to the bottom of the floor valve. The adapter is positioned on top of the reactor fuel transfer port and extends from the reactor head to the bottom of the floor valve, which is located at the elevation of the reactor-containment-building operating floor. The adapter serves as an extension of the reactor cover gas containment and provides shielding when irradiated core components are removed from the reactor. It also guides cooling air from an air blower to a cooling insert inside the adapter.

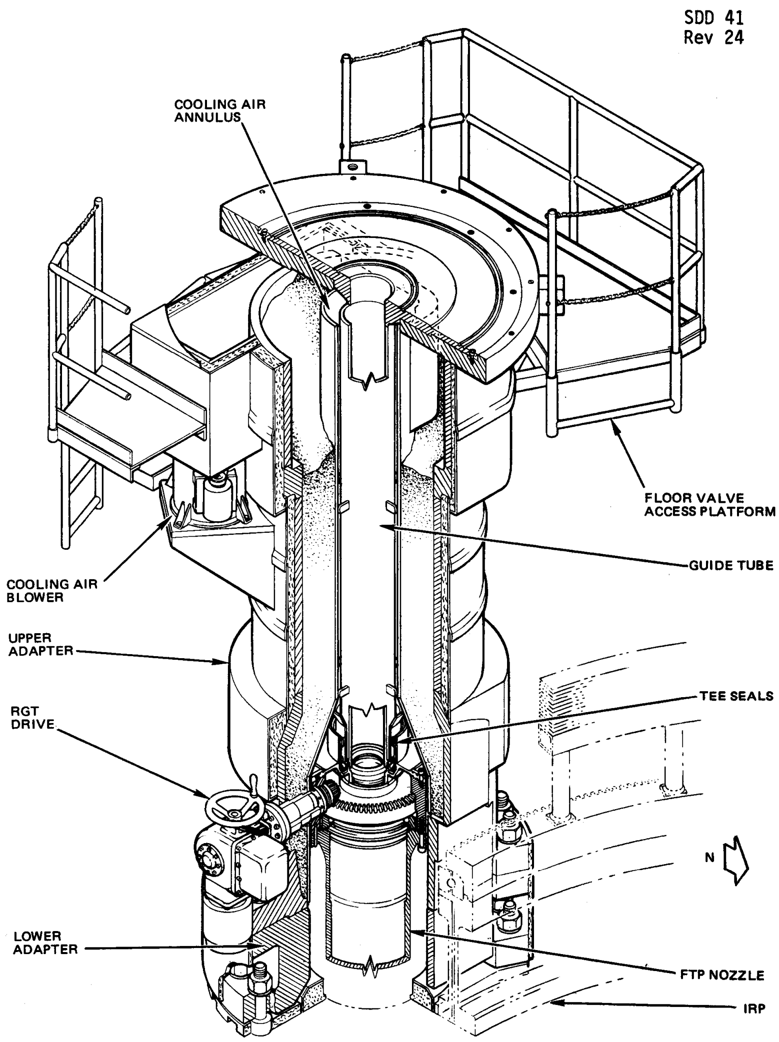

Fig. B.11. Reactor fuel transfer port adapter assembly.

B.3.8 Rotating Guide Tube

The function of the rotating guide tube (Fig. B.12) is to provide a means by which the EVTM can insert a new core assembly in a CCP into one of two fixed in-vessel transfer positions and remove an SNF assembly in a CCP from the other transfer position, without requiring the EVTM to decouple from the reactor vessel and recouple to the reactor vessel over a second refueling port. Elimination of this decoupling and recoupling procedure results in a savings of several days of reactor downtime per refueling. The rotating guide tube is a straight tube with an eccentric lower end that is mounted on the top of the reactor fuel transfer port nozzle. It remains in the reactor during reactor operation and operates only during reactor shutdown for refueling. The rotating guide tube rotates about 180^{o} to locate the center of the eccentric lower section over either of two adjacent transfer positions. Any adjacent two of the five in-vessel transfer positions can be used by appropriate positioning of the large rotating plug.

B.3.9 Ex-Vessel Storage Tank

EVST capabilities include (1) preheating new core assemblies, (2) storing new and irradiated core assemblies under sodium, (3) cooling irradiated core assemblies, (4) containing the argon cover gas, (5) providing structural support and physical separation of fuel assemblies to maintain subcriticality, and (6) providing radiation shielding.

The EVST (Fig. B.13) is a sodium-filled storage facility with a two-tier rotating turntable and a fixed closure head with fuel transfer ports. The EVST is located in the reactor service building between the EVTM gantry rails in a nitrogen-gas-filled concrete vault. The EVST performance characteristics are provided in Table B.3. Major EVST components and their primary functions are listed in Table B.4.

The storage vessel and turntable are top supported. The guard tank is bottom supported. The closure head and turntable loads are transmitted to the storage vessel flange, which in turn, transmits the loads through the storage-vessel support structure to the vault concrete. No seismic restraint is required between the turntable and storage vessel.

The storage vessel is surrounded by the guard tank as a safety measure to protect against any sodium leakage from the storage vessel, so that cooling of irradiated assemblies can be maintained. The volume between the guard tank and storage vessel is small enough to maintain a minimum safe sodium level above the core assemblies stored in the upper tier of the vessel, in the event of a gross vessel failure, and large enough to allow for in situ weld inspection of the storage vessel exterior surface and the guard tank inner surface by an inserted movable in-service inspection device. The outside of the guard tank is insulated and has electrical heaters.

The core-assembly storage positions are cylindrical tubes, restrained and supported by grid plates in the turntable. Each storage tube holds two CCPs, one above the other. A maximum of 650 core assemblies can be stored in 9 circular rows in the EVST. A total of 668 positions are provided in the EVST design, but 18 of these are reserved for EVST neutron absorbers and are not available for core assembly storage.

Nine storage positions in various rows of the upper tier and nine corresponding positions in the lower tier contain B_4C-filled neutron absorber assemblies and are inaccessible for fuel storage. The purpose of the absorber assemblies is to limit the value of k_{eff} for the EVST to ≤ 0.95.

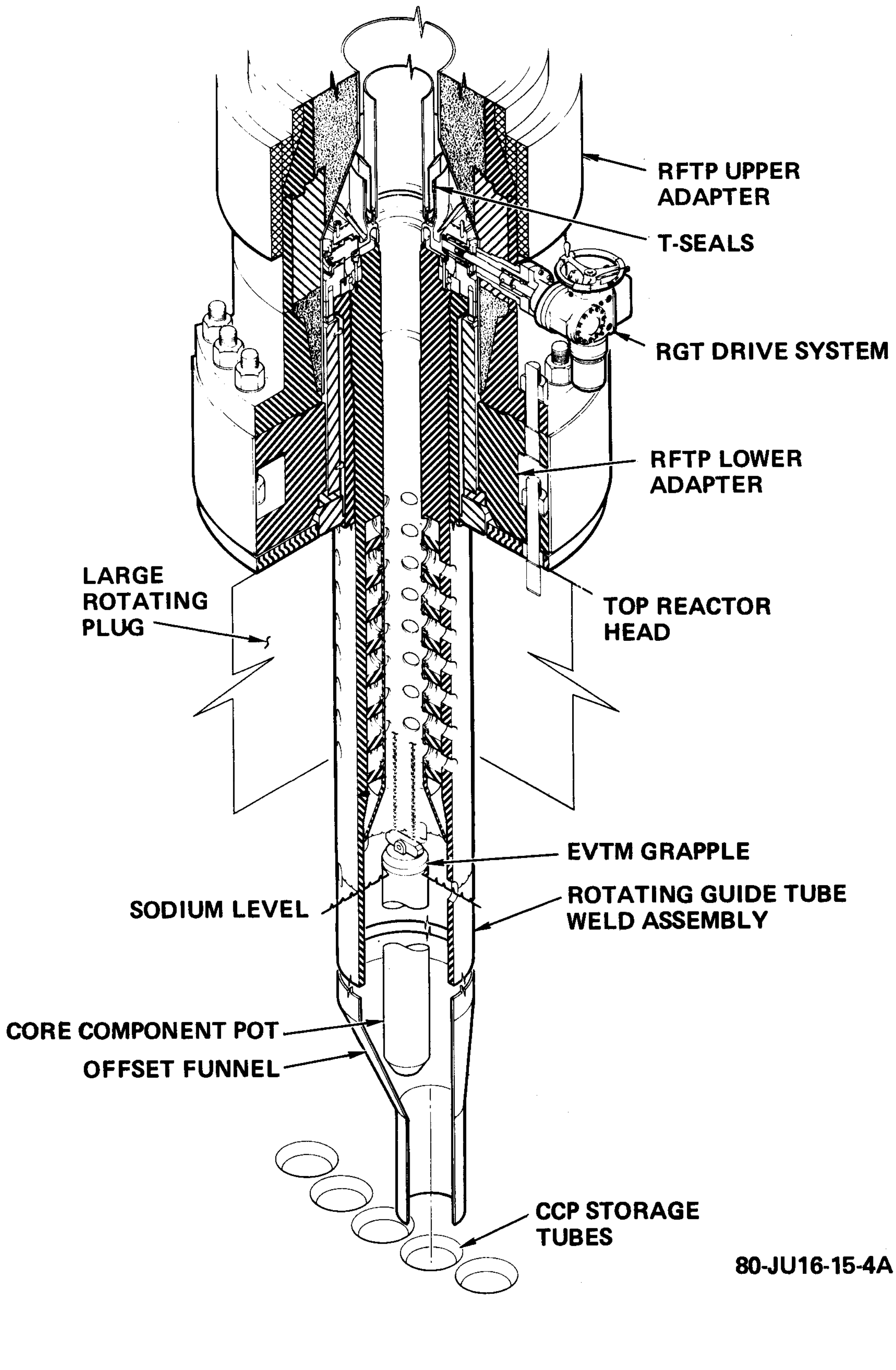

Fig. B.12. Rotating guide tube.

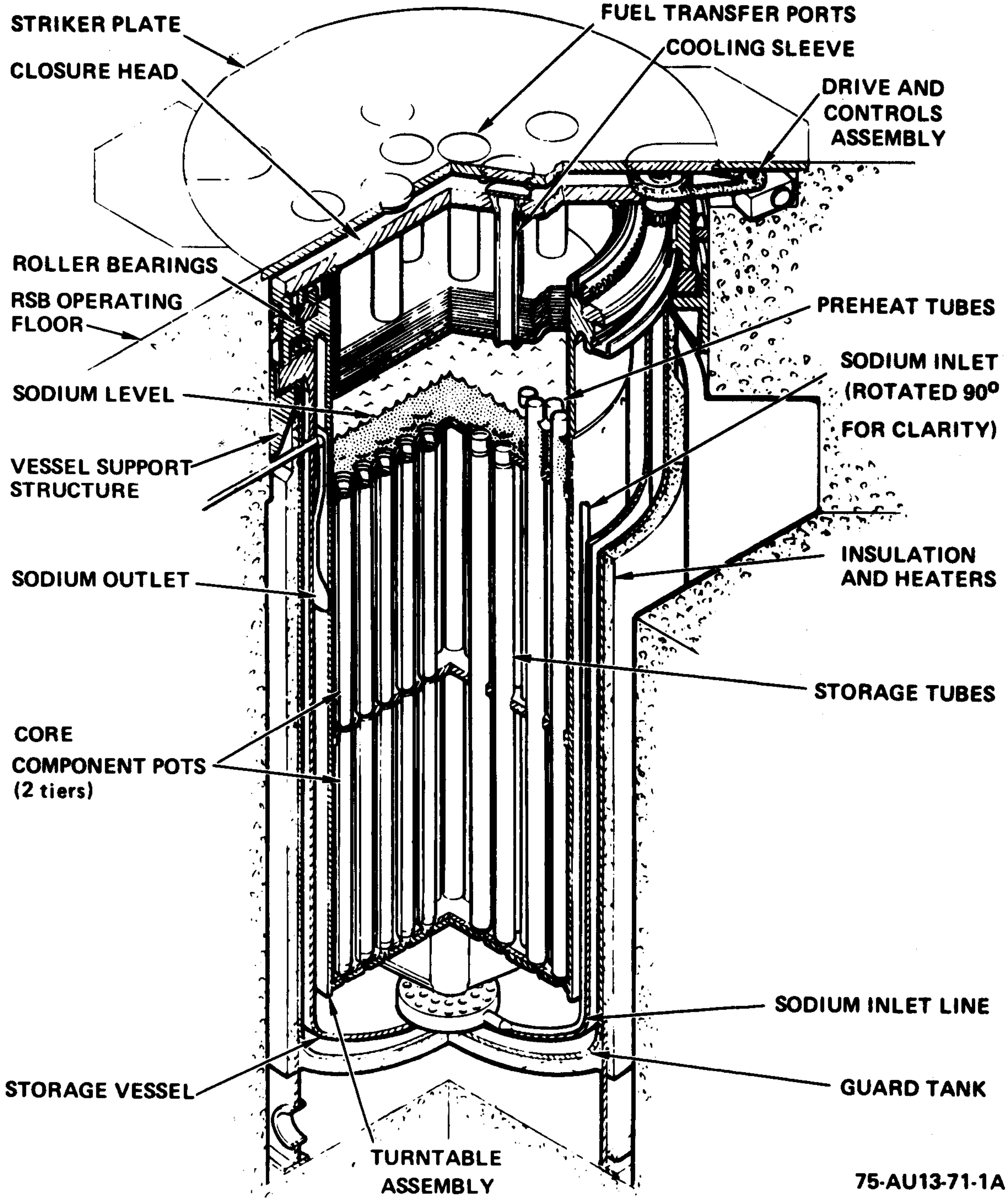

Figure B.13. Ex-vessel storage tank.

Table B.3. EVST performance characteristics

Number of storage positions	650
Design heat load, kW	1800
Sodium inlet temperature, °F/°C	~400/~204
Sodium flow rate, gpm	~400
Storage vessel design temperature, °F/°C	650/34
Containment-boundary design pressure, psig	±15

Table B.4. EVST major components and functions

Component name	Function
Storage vessel	Contain liquid sodium
Guard tank	Contain liquid sodium and maintain safe level
Closure head	Provide sealing of the cover gas and radiation shielding for reactor-service-building operating floor
Turntable	Support CCPs and maintain Subcriticality of fuel assemblies by physical separation
Support structure	Support storage vessel, guard tank, and interfacing refueling equipment loads and transfer these loads to the vault concrete
Drive controls	Provide for rotational movement and positioning of the turntable

The EVST also contains 24 dry, argon-gas-filled preheat tubes, 6 in each of the outer 4 rows of the upper tier, where new core assemblies are placed for preheating prior to immersion in the sodium. Core assembly temperatures increase at a slow rate when placed in these gas-filled positions, since heat transfer from the thimble wall to the assembly by both gas conduction and thermal radiation is slow. Under normal conditions, the preheat tubes reduce the effective storage capacity of the EVST. In addition, each row has 1 vacant storage position in each tier for ease of shuffling, reducing the effective storage capacity by another 18 positions, to a usable total of ~600. In the event that the entire storage capacity is needed (e.g., in case of a complete core unloading), the preheat tubes can be removed and the vacant positions filled.

Access to the storage positions is obtained through nine fuel transfer ports in the closure head, which are located within the range of the EVTM. A cold-wall cooling sleeve is provided around each fuel transfer port in the EVST to provide emergency cooling in the unlikely event that a CCP containing a spent fuel assembly should become immobilized while passing through the port. A blower is also provided to blow building air through the cooling sleeve in such an event.

The turntable is rotated to gain access to each storage position in each row. Rotation is by means of a bearing and two drive assemblies. Pinion gears are driven by speed reducers through a roller chain, and the pinion gears drive the bull gear which is attached to the turntable.

The EVST has three independent and redundant heat removal systems. Any one system is capable of removing the maximum design decay heat.

EVST shielding above the sodium pool consists of 22 in. of steel including the closure head, plus 2 in. of borated polyethylene neutron shielding attached under the striker plate. This shielding will ensure that the radiation level above the striker plate will not exceed 0.2 mrem/hr under the design basis conditions.

B.4 REFERENCES

CRBRP, 1974a, *Clinch River Breeder Reactor Plant Reference Design Report, Vol. 1*, June.

CRBRP, 1974b, *Clinch River Breeder Reactor Project: 1974 Technical Progress Report*.

CRBRP, 1983, *Clinch River Breeder Reactor Plant System Design Description, Reactor Refueling System*, SDD-41, Rev. 27, July.

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