R MODULES

Reactors and Transmuters

Preface for R Modules

The R modules are incrementally being updated to reflect the estimation methodologies discussed in the Cost Estimating Methods chapter of the main report. In this edition, the R1 module has been updated using a detailed combination of Engineering and Extrapolation from Actuals method. The R2 module has been updated using a Parametric estimation method documented in the supporting document 2017 SD6 (Parametric Approach for R Modules). With the exception of the pressurized Heavy Water Reactor (PWHR or "CANDU" Reactor: Module R5), most of the other reactor types are less mature, and the planned approach is to update them in subsequent editions using either this same parametric approach or an Analogy approach using partial correlation coefficients that is under development as discussed in the supporting document 2017 SD3 (Cost Correlations).

Module R1

Light Water Reactors

Module R1

Light Water Reactors (LWRs)

R1.MD SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Complete technical revision of overnight capital cost. O&M escalated from AFC-CBR 2015.
- **Estimating Methodology for latest (2017 AFC-CBR) technical update from which this 2017 update escalated:** Past construction experience, along with on-going construction experience, was evaluated to look for the best experience attributes. This was done in order to assess the overnight cost **without** the significant cost overruns due to the many factors that are **not** part of the cost of well-built reactor, which is the result of a well-executed construction project.

R1.RH REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2004 as Module R1.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2017
- **New technical/cost data which has recently become available and will benefit next revision:**
  - A recent report from the Breakthrough Institute (Lovering 2016) includes some data on Far Eastern NPP costs.

R1-1. BASIC INFORMATION

The reactor is the central facility of the overall energy system and is supported by the nuclear materials initially processed in the fuel cycle "front end," "burned" in the reactor, and finally dispositioned or recycled in the overall fuel cycle "back end." This section deals with the light water moderated class of "thermal" reactors; i.e. reactors in which the average neutron energies are in the thermal or "slow" range (~0.025-eV), and for which moderators of low atomic number are required. In thermal reactors, the moderators most commonly used are light water (this Module R1), heavy water (Module R5), or graphite (carbon) (Module R3). All operating commercial reactors in the U.S. are of this thermal type, all being pressurized water reactors (PWRs) or boiling water reactors (BWRs), approximately two to one in deployment ratio, respectively. As a group, these U.S. plants are called light water reactors (LWRs). Their name distinguishes them from heavy water reactors (HWRs) such as those used predominantly in Canada (CANDU: Module R5), and gas-cooled reactor s (Module R3) such as those used predominantly in the UK, which use a graphite moderator.

LWRs come in two basic types, the Pressurized Water Reactor (PWR) and the Boiling Water Reactor (BWR). Figure R1-1 below shows the PWR concept, which has two light water flow loops. The primary water coolant is at high pressure and remains a liquid at temperatures around 315 C. Heat is transferred to a secondary loop where steam produced at around 290 C drives the turbine generator

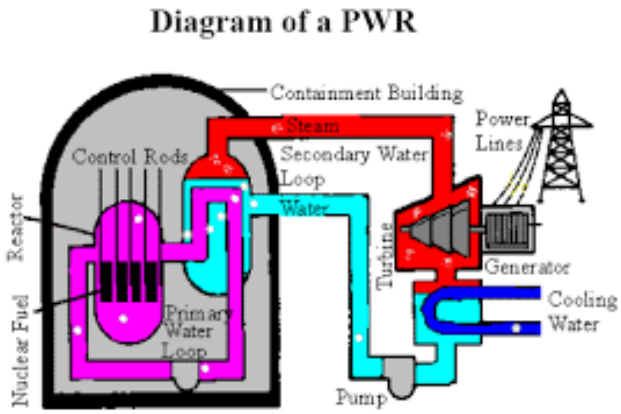

Figure R1-1. Pressurized Water Reactor Concept.

In the BWR concepts shown in Figure R1-2 below, the water in the core exists in both the liquid and vapor phase. Steam from the top of reactor vessel drives the turbine-generator in a single coolant loop.

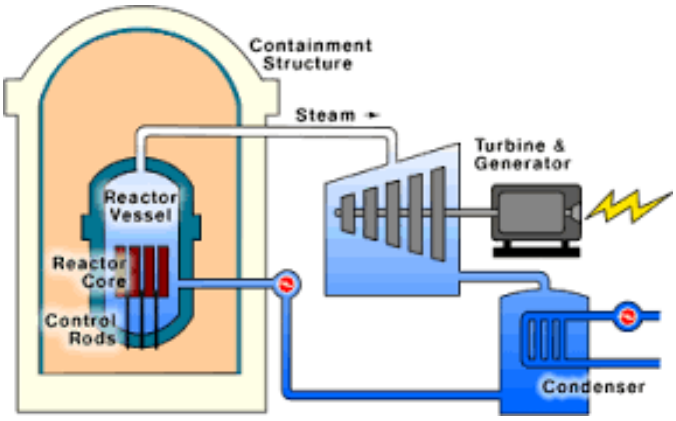

Figure R1-2. Boiling Water Reactor Concept.

Most LWRs in the world are the PWR type; however, both types have enjoyed excellent performance and reliability worldwide.

The predominant product from LWRs is electricity. However, the heat generated (in the form of steam) can also be used for industrial applications such as district heat, process heat, or water desalination. Capacities of existing U.S. thermal reactors vary from a few hundred megawatts of electrical power per unit to around 1,600 MWe per unit. A nuclear power plant may actually have more than one unit (reactor) on the same site. The Palo Verde plant in Arizona has three reactors on one site. The fuel cycle cost for a reactor is just one of the four main components of the busbar levelized unit electricity cost (LUEC) from a nuclear power plant. ("Busbar" cost refers to the fact that the electricity cost is measured at the reactor plant boundary connection on the primary side of the switchyard transformer and does not include distribution [transmission] or other utility overhead costs.) The LUEC is usually expressed in mills/kWh or $/MWh; the value is the same in these two units. (One mill=1/1,00 0th of a dollar or 0.1 cents). This and other economics-related definitions are described in the *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* (G4-EMWG 2005). The four components of the LUEC are:

1. Capital component: recovery of reactor capital plus financing costs. The capital component includes all "up-front" costs prior to commercial operation, including: design, licensing, construction, project management, ownership costs, interest during construction, and reactor start-up (commissioning).

This component of the LUEC also includes the returns to the investors made during plant operations, such as the interest portion of capital recovery.

2. Operations and maintenance component: annual nonfuel costs including manpower, nonfuel consumables, and overheads. Manpower costs for refueling outages are usually captured in this category. Replacements for major capital items not related to life extension, such as steam generators, can also be placed in this category.

3. Fuel cycle component: the sum of the relevant costs for the needed fuel cycle steps (modules) converted to mills/kWh or $/MWh unit costs. Models such as G4-ECONS can perform this sometimes complex calculation (G4-EMWG 2006), which involves both unit costs for fuel cycle steps and fuel cycle material balances. Depending on the utility, accounting practices, carrying charges (interest) on stored fuel, and fuel cycle materials undergoing processing are sometimes assessed to this category.

4. Decontamination and decommissioning (D&D) component: usually covered by an escrow or sinking fund accumulated to cover D&D costs for the reactor at its end of life. The calculation of the levelized annual payments to this fund over the operational life of the reactor is described in G4-EMWG 2005.

These four components are ranked from top to bottom with the highest contributors to LUEC at the top. Table R1-1 shows the projected contributors to LUEC for an "Nth-of-a-kind" (NOAK) Generation III PWR design (ABB-CE System 80+). The example table was generated by the G4-ECONS model (2006). All values in the table are in constant (unescalated) 2001 dollars, and fuel cycle costs are based on the lower values of fuel cycle materials and services in that year. The discount (interest) rate is typical of a lower risk, highly-regulated financial environment. Since we are dealing with constant dollars, a "real" or uninflated discount rate is used. This older 2001 example was provided because detailed PWR cost input data was available for G4-ECONS input and it could be benchmarked against other LUEC models. The more current (2009-2017) reactor cost environment (higher construction and fuel costs) is discussed later in this section.

Table R1-1. Projected breakdown (in 2001 $) of electricity cost (LUEC) for a 1,300 MWe Generation III thermal reactor as calculated by G4-ECONS (G4-EMWG 2006).

Summary of Model Results		
Discount Rate =	5.00%	
	Annualized Cost in $M/Year	Mills/kwh or $/MWh
Capital (Including 1st Core and Financing)	158.52	17.40
Operations Cost	78.47	8.61
Fuel Cycle - Front End	29.07	3.19
Fuel Cycle - Back End	9.90	1.09
D&D Sinking Fund	2.48	0.27
TOTAL LUEC	**278.44**	**30.56**

The capital component is always the largest of these costs, which is different than fossil-fuel electricity generation sources, such as oil, natural gas, or coal, where recurring fuel purchase costs can be predominant and also unstable—subject to wide market price fluctuation. The low fuel cycle cost is one of the advantages of nuclear power and is due, in part, to the fact that nuclear fuel (uranium or plutonium) delivers nearly one-million times the energy per unit mass than chemical fuel sources such as fossil fuels

(higher energy density). The high capital cost of nuclear power is partly because of the need to safely confine the highly energetic and radioactive reactor core and prevent radioactive materials from escaping to the environment or harming plant workers and the public. Because of the possibly catastrophic consequences of a nuclear accident, nuclear power plants must be constructed to much more stringent safety and quality standards than those for fossil power plants, but stringent regulation also contributes to higher costs. Massive amounts of steel and concrete with the associated installation labor and quality assurance are required for nuclear power plants.

The most interesting and useful cost figure of merit associated with a reactor project is the "specific" capital cost, which is the cost of planning, licensing, designing, constructing, and starting up the reactor divided by the power capacity. It is usually expressed in $/kilowatt electric ($/kWe.) One must be careful to specify whether the capital cost includes financing (interest) costs and other ownership costs. If not, the capital cost is called the "overnight" cost—the cost if the plant could be built "overnight" and not encounter any interest costs. The total capital cost (TCC) includes interest during construction, which can be a significant percentage of the overnight cost because of the multi-year construction period. (This sensitivity to financing assumptions is examined later in this section.) The following discussion deals mostly with the "overnight" expression of the specific capital cost because it is most dependent on the reactor technology and construction efficiency.

In the paragraphs above from the 2009 AFC-CBR the generic technical and economic considerations associated with LWR were described in detail. In the 2012 AFC-CBR the scope of Module R1 was changed somewhat. This type of reactor is still the predominant type of reactor in the US and in the world; however, the 2012 AFC-CBD Module deals with the "large" or "gigawatt class" version of this reactor type. (Module R4's 2012 update dealt mainly with the small modular LWR.) It should be kept in mind that even newer large LWRs are somewhat modular in nature, since many reactor subsystems can be assembled in factories and shipped to the reactor site. This is indeed the case for the Vogtle and Summer AP-1000 PWRs, either recently or now under construction in Georgia and South Carolina respectively, where many modules are built in Lake Charles, Louisiana.

R1-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

Thermal reactors all use uranium oxide (see Module D1-1) or mixed oxide (see Module D1-2) fuel in some form. LWRs and HWRs use pelletized ceramic fuel clad in zirconium or zirconium alloy rods. The rods are bundled into fuel assemblies that are inserted into the reactor prior to startup. U.S. concept gas-cooled thermal reactors have uranium oxycarbide particle fuel in the form of tiny beads that are coated with heat and diffusion-resistant coatings. Module D1-3 describes this "TRISO" type of fuel, which is embedded in a graphite matrix. The internal heat generated by fission of the U-235 and Pu-239 in the fuel is removed by flowing coolant and transferred by pumps, heat exchangers, and steam generators to a rotating turbine that generates electricity. Figures R1-1 and R1-2 show this schematically. Because thermodynamic cycles (Rankine cycle for LWRs and Brayton cycle for direct cycle GCRs) are involved, most of the heat energy is rejected to the environment, as is true of all "thermal" (in the thermodynamic rather than neutronic use of this term) power plants using fossil or nuclear fuel. The ratio of electric power generated to total heat generation is the thermodynamic efficiency. Other important reactor performance parameters are:

The capacity factor: the number of effective full power hours divided by the total hours in the year. This factor is lowered by planned or unplanned outages. Outages are planned for refueling and scheduled maintenance, normally during times of lower power demand such as fall or spring. Today's typical U.S. LWR enjoys a capacity factor of over 90%.

The fuel burnup: expressed in (thermal) megawatt-days per metric ton of heavy metal, this figure-of-merit designates the amount of energy that can be extracted from a unit mass of fuel. The accumulation of nuclear poisons (neutron-absorbing nuclides) and degradation of fuel materials (cladding, pellet integrity, etc.) limit the lifetime of a fuel assembly in the reactor.

The vintage of reactor technology used is referred to as its "generation." Early prototype and small commercial (a few hundred MWe) units are designated as Generation I. The later and larger units built in the 1960s, 1970s, 1980s, and 1990s are called Generation II. The advanced LWRs and evolutionary design units being built today in the Far East and proposed for construction in the U.S. by 2020 are Generation III or III+ units. These units may incorporate passive safety features. Generation IV reactors are those proposed for deployment after 2020 that may use advanced safety features, incorporate waste minimization, and have additional economics-enhancing and proliferation-resistant features. They are the subject of several extensive international research and development (R&D) programs involving several nations and six technology concepts.

Another reactor category for which interest is growing is that of small to medium reactors (SMRs). (The IAEA defines a small reactor as 300 MWe or less, and a medium-sized reactor as 300 to 700 MWe.) The smaller reactors of this type are sometimes referred to as "grid-appropriate" reactors (GARS) or "deliberately small" reactors (DSRs). The market for such reactors would be for localities or utilities, which cannot afford the high expense of a large reactor, and may not have a power system grid able to accommodate the large reactor. Both fast and thermal neutron type reactors have design candidates in this category. This reactor type and its cost was discussed in detail in Module R4 of the 2009 & 2012 updates AFC-CBRs but is not included in the 2017 version.. Since "modularity" is an issue generic to all types of reactors its discussion has been covered in a chapter of the Main Report for this 2017 AFC CBR.

The following describes the commonly used code of accounts that is used in the breakdown of the cost of reactors. This breakdown is from EEDB 1988 (ORNL 1988b):

- **Structures and Improvements (Account 21):** This account includes the on-site surface buildings, structures, related subsurface foundations, tunnels that house and support all equipment, components, piping, ducting, and wiring, except for the foundations for individual plant machinery or the buildings and foundations heat rejection systems. Included are site improvements, such as excavation, grading, roadways and railroads, substructure and superstructure details and architectural features. Also included are equipment and piping for the heating, ventilating and air conditioning systems, piping for the roof, floor and sanitary drains, and equipment for the lighting and service power systems.
- **Reactor or Boiler Plant Equipment (Account 22):** This equipment includes the reactor, reactor safety systems, fuel storage systems, and radioactive waste handling systems. Also included are the interconnecting piping systems, structural supports for equipment, and necessary instrumentation and control systems.
- **Turbine Plant Equipment (Account 23):** This account includes the power conversion system equipment including the turbine-generator unit, condenser, systems to purify and return the condensate to the reactor or boiler plant (condensate and feedwater systems), elevated turbine-generator pedestal, main vapor piping system, auxiliary support systems, interconnecting piping systems, structural supports for equipment, and necessary instrumentation and control systems.
- **Electric Plant Equipment (Account 24):** This account includes the systems and equipment required to deliver the generated electric power to the off-site transmission system, provide auxiliary electric power for all power plant equipment and auxiliaries, and provide standby power for safety systems for nuclear power plants or emergency backup power for selected systems for fossil power plants. Included are the cable and raceways for all power, control and instrumentation systems, structural supports for equipment, generator control system equipment and plant grounding, lightning protection, freeze protection and cathodic protection equipment. Although building lighting and service power equipment are included in Account 21, the equipment for the distribution of power to these systems is included in this account.

- **Miscellaneous Plant Equipment (Account 25):** This account includes the auxiliary mechanical and electric equipment required for normal power plant start-up, operation and maintenance including the transportation and lift equipment (cranes), equipment in the air, water and steam service system, auxiliary boiler, fire detection and protection systems, communication system, non-radioactive waste water treatment system, various plant monitoring systems, miscellaneous furnishings and fixtures, and necessary interconnecting piping systems and structural supports for this equipment.
- **Main Condenser Heat Rejection System (Account 26):** This account includes the equipment and associated structures and piping that dispose of the heat rejected by the power plant and provide make-up water to the power plant including the cooling towers and the structures, equipment and interconnecting piping systems for obtaining and pretreating the plant make-up water.
- **Construction Services (Account 91):** This account includes the temporary structures and facilities, janitorial services, maintenance of temporary facilities, guards and security, roads, parking lots, laydown areas, temporary electrical, heat, air, steam and water systems, general cleanup, and related items and activities; The rental and/or purchase of construction equipment, small tools and consumables (fuel, lubricants, etc.), as well as maintenance of construction equipment. Insurance and taxes related to craft labor, such as Social Security taxes and state unemployment taxes, workmen's compensation insurance, and public liability and property damage insurance; Permits, Insurance and Local Taxes and Builders all-risk insurance.
- **Engineering and Home Office Services (Account 92):** This account includes the Home Office Services: the salaries of personnel, direct payroll-related costs (DPC), overhead loading, expenses and related fees associated with the engineering and design (both home office and field), procurement and expediting activities, estimating and cost control, engineering planning and scheduling, and reproduction services, plus expenses associated with performance of the above functions (i.e., telephone, postage, computer use, travel, etc.); Home Office Quality Assurance (Nuclear Power Plants): the salaries, DPC, overhead loading and expenses (e.g., travel) associated with the services of home office quality assurance engineers and staff personnel including reviews, audits, vendor surveillance, and other activities as required for design and construction of the nuclear safety-related portion of the facility. Home Office Construction Management: the salaries, DPC, overhead loading and expenses associated with the services of the construction manager and his assistants including construction planning and scheduling, construction methods, labor relations, and utilization of safety and security personnel.
- **Field Supervision and Field Office Services (Account 93):** This account includes the Field Office Expenses: the costs associated with purchase and/or rental of furniture and equipment (including reproduction), communication, postage, stationery, other office supplies, first aid and medical expenses. Field Job Supervision: the salaries, DPC, overhead loading, relocation costs and fees associated with the resident construction superintendent and his assistants, craft labor supervisors, field accounting, payroll and administrative personnel, field construction schedulers, field purchasing personnel, warehouse personnel, survey parties and clerical personnel; Field Quality Assurance/Quality Control: the salaries, DPC, and overhead loading associated with personnel located at the job site engaged in equipment inspection, required documentation of safety-related equipment (nuclear power plants only), inspection of construction activities and construction training meetings; Plant Startup and Test: the salaries, DPC, overhead loading, and miscellaneous related expenses associated with preparation of start-up and plant operation manuals and test procedures, direction and supervision of testing of equipment and systems as the plant nears completion and direction of start-up of the facility. (Costs of craft labor required for start-up and testing activities are included in the appropriate Direct Cost line items.)

R1-3. PICTURES, DIAGRAMS, AND DEPLOYMENT STATUS

The "Generations" concept as applied to reactors is explained in Figure R1-3.

To further the advancement of Generation III and III+ reactors in the U.S., the U.S. Department of Energy (DOE) instituted the NP-2010 program. The NP-2010 program focused on reducing the technical, regulatory, and institutional barriers to deployment of new nuclear power plants based on expert recommendations documented in *A Roadmap to Deploy New Nuclear Power Plants in the United States by 2010* (DOE-NE 2001).

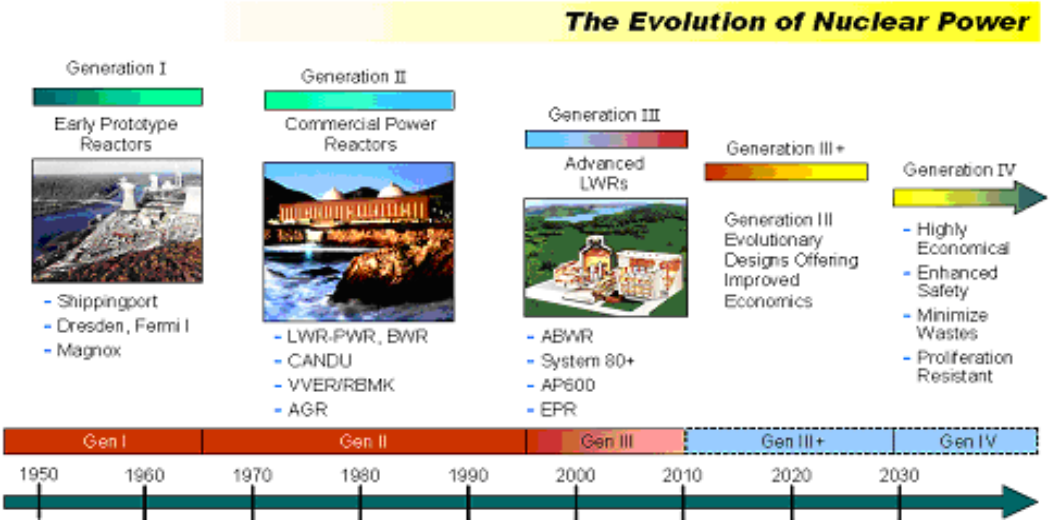

Figure R1-3. Nuclear power evolution by "generations" (DOE 2001).

The technology focus of the Nuclear Power 2010 program was on Generation III+ advanced LWR designs, which offer advancements in safety and economics over the Generation III designs certified by the Nuclear Regulatory Commission (NRC) in the 1990s. To enable the deployment of new Generation III+ nuclear power plants in the United States in the relatively near future, it is essential to completely develop the first-of-a-kind Generation III+ reactor technology and demonstrate the untested federal regulatory and licensing processes for the siting, construction, and operation of new nuclear plants. DOE utilizes competitive procurement processes and conducts program activities in cost-share cooperation with industry. DOE has initiated cooperative projects with industry to develop the business case for new nuclear power plants, to obtain NRC approval of three sites for construction of new nuclear power plants under the Early Site Permit (ESP) process, to support completion of Generation III+ design engineering work, to resolve generic COL regulatory issues, and to support the NRC review of COL applications. The COL process is a "one-step" licensing process by which nuclear plant public health and safety concerns are resolved prior to commencement of construction, and NRC approves and issues a license to build and operate a new nuclear power plant.

The Energy Policy Act of 2005 also included investment stimuli for new nuclear power plants (NPPs). These included:

- Federal loan guarantees that cover up to 80% of the project cost
- Production tax credits for 8 years of $18/MWh for up to 6,000 MWe of capacity, limited to $750,000,000 per year
- Federal standby support (to cover some of the economic damages from regulatory delays)
  - $2B of risk coverage for first six plants

- Coverage for delays resulting from licensing or litigation.

The intent of these incentives was to make investors (Wall Street) more likely to finance the high up-front costs required for a reactor project in the middle of the first decade of the 21st century. Utility interest in these incentives was intense as was evidenced by the number of projects that announced to the NRC their intent to pursue a COL. Table R1-2 lists the projects announced as of Summer 2009. Since 2009 most of these projects have been cancelled or deferred for economic reasons. Only Vogtle is still under construction, and for economic and market reason its completion is uncertain as of the end of 2017.

Table R1-2. New plant table as compiled by the nuclear regulatory commission in 2009.

Proposed New Reactor(s)	Design	Applicant
Bell Bend Nuclear Power Plant	U.S. EPR	PPL Bell Bend, LLC
Bellefonte Nuclear Station, Units 3 and 4	AP1000	Tennessee Valley Authority (TVA)
Callaway Plant, Unit 2	U.S. EPR	AmerenUE
Calvert Cliffs, Unit 3	U.S. EPR	Calvert Cliffs 3 Nuclear Project, LLC and UniStar Nuclear Operating Services, LLC
Comanche Peak, Units 3 and 4	US-APWR	Luminant Generation Company, LLC (Luminant)
Fermi, Unit 3	ESBWR	Detroit Edison Company
Grand Gulf, Unit 3	ESBWR	Entergy Operations, Inc. (EOI)
Levy County, Units 1 and 2	AP1000	Progress Energy Florida, Inc. (PEF)
Nine Mile Point, Unit 3	U.S. EPR	Nine Mile Point 3 Nuclear Project, LLC and UniStar Nuclear Operating Services, LLC (UniStar)
North Anna, Unit 3	ESBWR	Dominion Virginia Power (Dominion)
River Bend Station, Unit 3	ESBWR	Entergy Operations, Inc. (EOI)
Shearon Harris, Units 2 and 3	AP1000	Progress Energy Carolinas, Inc. (PEC)
South Texas Project, Units 3 and 4	ABWR	South Texas Project Nuclear Operating Company (STPNOC)
Turkey Point, Units 6 and 7	AP1000	Florida Power and Light Company (FPL)
Virgil C. Summer, Units 2 and 3	AP1000	South Carolina Electric & Gas (SCE&G)
Vogtle, Units 3 and 4	AP1000	Southern Nuclear Operating Company (SNC)
William States Lee III, Units 1 and 2	AP1000	Duke Energy

R1-4. MODULE INTERFACES

The reactor receives fuel assemblies from the fuel fabrication plant (see Module D1-1) for uranium oxide-fueled thermal reactors, (see Module D1-2) for mixed oxide fueled thermal reactors, or (see Module D1-3) for gas-cooled thermal reactors. Module D1-7 covers the fuel supply for CANDU HWR reactors, and Module D1-8 covers thorium-based fuel in thermal reactors, but mainly for those of Russian design (VVERs).

After irradiation, fuel assemblies are stored in an onsite pool. At some point, the fuel assemblies might be moved to storage casks for onsite or offsite storage (see Module I). Direct transfer to an aqueous reprocessing facility is also possible (see Module F1). Module G2 considers the costs of conditioning and repackaging spent fuel if required before transport.

R1-5. SCALING CONSIDERATIONS

It is not clear if economies of scale can be proven to exist for LWRs, or if they even become dis-economies of scale after a certain size, based on the historical US construction experience. It is generally possible to obtain scaling factors for individual components, but in general it is not recommended to

simply scale costs for a certain power reactor to a different power level using a power exponent, until more definite proof of economies of scale can be demonstrated.

R1-6. COST BASES, ASSUMPTIONS, AND DATA SOURCES

After a description of historical construction costs in the U.S. and France, and the introduction of a framework to interpret the construction cost escalation and cost overruns, a summary is provided with recommended expected values and probability density functions for the "engineering" and "non-engineering" construction costs.

R1-6.1. The Historical U.S. Reactor Construction Cost

A brief history and cost data of the U.S. LWR reactor construction experience is provided in this section. Historical information for other countries can be also provided in this Section, as the information is collected and considered reliable/defensible enough, in successive CBR updates.

The Atomic Energy Commission's (AEC) Power Reactor Demonstration Program, which existed between 1955 and 1963, subsidized the construction of the first demonstration power reactors in the U.S.: a dozen 10-75 MW reactors that started service in the 1950s and three 200 MW-class reactors that started service in the early 1960s. The beginning of commercial nuclear energy in the U.S. was 1968, with the commercial operation of the first 500 MW class reactors (Haddam Neck (CT) and San Onofre (CA)).

Afterwards, a dozen "turn-key" plants started operations: these plants were sold at a fixed price by Westinghouse LLC and by General Electric Co. in order to demonstrate the commercial viability of nuclear energy. The first of these was the 650 MW Oyster Creek (ordered in 1963), while the almost identical Nine Mile Point 1 was ordered months later on a commercial (i.e. non turn-key) basis. The cost data for the first turn-key units are either unavailable, and/or unreliable, and therefore they are not included in the cost-database presented here. The following 15, mostly turn-key U.S. reactors built before 1971 are therefore excluded from the database:

- Connecticut Yankee (also known as Haddam Neck);
- San Onofre 1;
- Oyster Creek;
- Nine Mile Point 1;
- Ginna,
- Dresden 2 and 3,
- Point Beach 1 and 2,
- Millstone 1,
- Robinson,
- Monticello,
- Quad Cities 1 and 2,
- Indian Point 2.

The cost database shown in Table R1-3 is based on the data in (Koomey and Hultman, 2007) for 99 U.S. reactors, containing, among other things: reactor name, overnight cost, total capital investment in today's dollars and construction duration. In turn, (Koomey and Hultman, 2007) relied mostly on data from (Komanoff, 1981) and later updated, with the exception of the cost for Comanche Peak 1 and 2 (which were built by Texas Utilities), Seabrook (which was completed by a consortium of utilities in the NorthEast), and Watts Bar 1 (which built by Tennessee Valley Authority (TVA)).

For each of these, the complete time series of the construction expenditures was obtained by the authors of (Koomey and Hultman, 2007), and converted to constant-1987-dollars using the Handy-

Whitman escalation index, and then to 2004 using the general GDP index. Additionally, (Koomey and Hultman, 2007) obtained:

1. Construction duration data from an NRC database,
2. Electrical power level and capacity factors from an IAEA database; and
3. Electrical efficiency from the Global Energy Decision Database through the NEI.

The cost data were originally collected by the Federal Power Commission and later by the DOE's Energy Information Administration, and published through several years in the *Steam-Electric Plant Construction Costs and Annual Production Expenses* (DOE/EIA-0033 series). The methodology to convert "as expended" dollars to "constant dollars" of a given year, is detailed in *An Analysis of Nuclear Power Plant Construction Costs* (DOE/EIA-0485) (DOE, 1986). Additionally, the cost data for the following plants were originally obtained by Komanoff through direct communication with the utilities: Browns Ferry 3, Crystal River 3, Indian Point 3, Fitzpatrick, Davis Besse 1.

Figure R1-4 shows the specific overnight capital cost (in 2004 $/kWe) for the U.S. nuclear power plants as a function of construction start-year and end-year. The cost data can be updated to 2014 using the Handy-Whitman escalation index for nuclear power plants, or the general inflation rate: for example, the CPI can be found online at the U.S. Bureau of Labor Statistics web site (BLS, 2014). The cumulative change in CPI between 2004 and 2014 is 1.26.

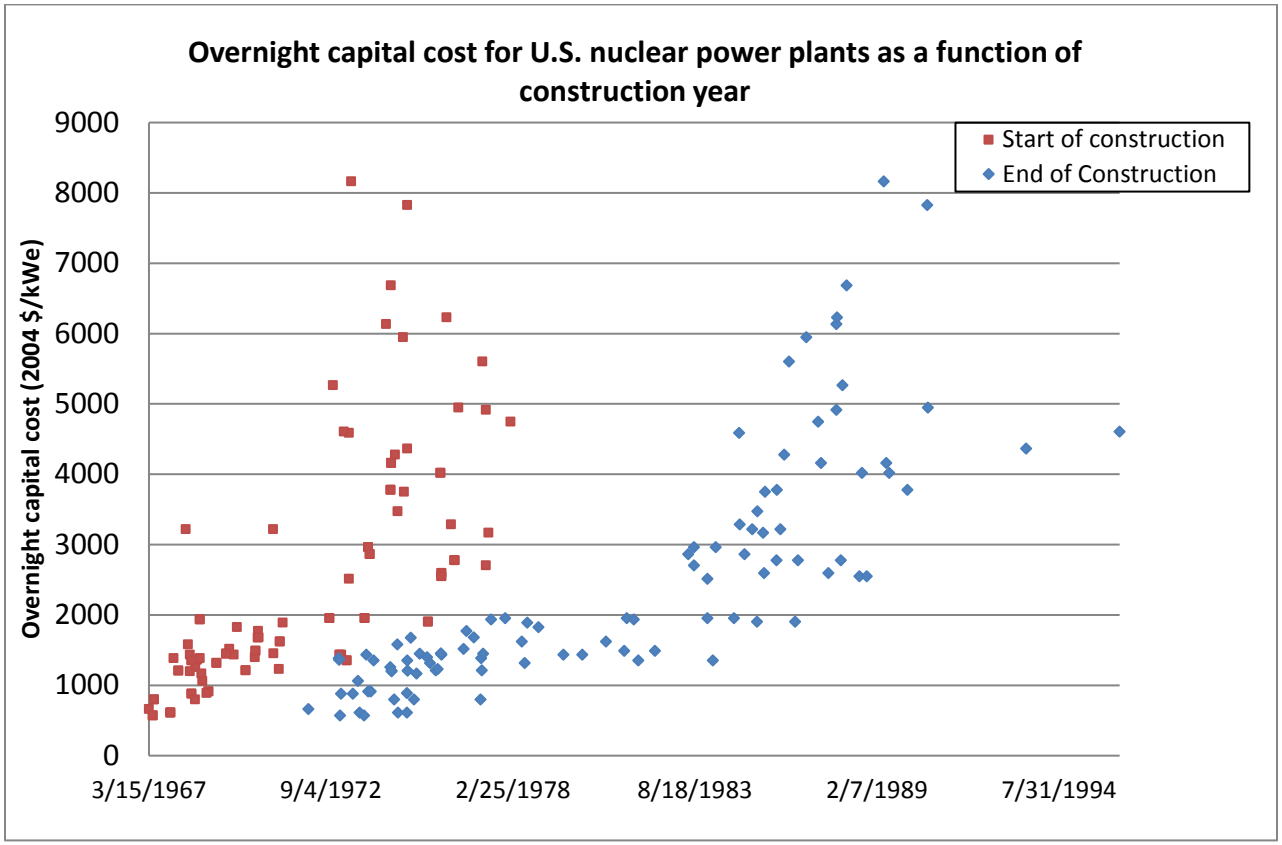

Figure R1-4. Overnight capital cost (in 2004 $/kWe) as a function of construction start year and end-year, from the U.S. reactor cost database of Appendix A. The distance between the red dots and the blue dots of the same vertical level is the construction duration.

The average U.S overnight construction cost was 2450 $\$_{2004}$/kWe (3086 $\$_{2014}$/kWe); the median construction cost was 1825 $\$_{2004}$/kWe (2300 $\$_{2014}$/kWe); Turkey Point 3 and 4 had the lowest overnight

cost at 571 $\$_{2004}$/kWe each (719 $\$_{2014}$/kWe); and Shoreham featured the highest overnight cost at 8165 $\$_{2004}$/kWe (10,288 $\$_{2014}$/kWe).

It can be observed from Figure R1-4 that a few of the earliest plants had an overnight capital cost lower than 1000 $\$_{2004}$/kWe, and were completed relatively quickly (often 4-5 years). Additionally, all the plants completed before 1983, did so for an overnight cost lower than 2000 $\$_{2004}$/kWe. Also, while cost escalation and lengthening construction times were already observable in the mid to late 1970s, after the TMI accident of March 1979 no new construction has been started, and costs escalated dramatically for the plants that reached completion. The framework proposed in Section R1-6.3 has been developed to explain these observed trends.

Table R1-3. U.S. Nuclear Plants Cost Table from (Ganda 2014)

Name of Reactor	MW		State	Start of construction	End of construction	Shutdown	Years for construction	Lifetime	Thermal Efficiency	Overnight capital cost (2004 $)	Total capital cost (2004 $)
Palisades	697	PWR	MI	3/15/1967	12/31/1971	12/31/2011	4.8	40	32.90%	664	745
Vermont Yankee	507	BWR	VT	12/12/1967	11/30/1972	11/30/2012	5	40	33.70%	1386	1561
Maine Yankee	879	PWR	ME	10/22/1968	6/29/1973	12/6/1996	4.7	23.4	32.50%	1064	1188
Pilgrim	672	BWR	MA	8/27/1968	12/2/1972	12/2/2012	4.3	40	33.50%	1361	1501
Surry 1	790	PWR	VA	6/26/1968	12/22/1972	12/22/2012	4.5	40	33.90%	881	978
Turkey Point 3	672	PWR	FL	4/28/1967	12/14/1972	12/14/2012	5.6	40	31.00%	571	656
Surry 2	793	PWR	VA	6/26/1968	5/1/1973	5/1/2013	4.8	40	33.90%	881	989
Oconee 1	851	PWR	SC	11/7/1967	7/15/1973	7/15/2013	5.7	40	32.80%	611	704
Turkey Point 4	673	PWR	FL	4/28/1967	9/2/1973	9/2/2013	6.4	40	31.00%	571	671
Prairie Island 1	511	PWR	MN	6/26/1968	12/16/1973	12/16/2013	5.5	40	31.80%	1352	1546
Zion 1	1069	PWR	IL	12/27/1968	10/19/1973	2/21/1997	4.8	23.3	32.50%	912	1023
Fort Calhoun	478	PWR	NE	6/8/1968	9/26/1973	9/26/2013	5.3	40	32.10%	1435	1632
Kewaunee	521	PWR	WI	8/7/1968	6/16/1974	6/16/2014	5.9	40	31.00%	1259	1457
Cooper	764	BWR	NE	6/6/1968	7/2/1974	7/2/2014	6.1	40	31.80%	1199	1397
Peach Bottom 2	1078	BWR	PA	2/1/1968	7/2/1974	7/2/2014	6.4	40	32.40%	1208	1422
Browns Ferry 1	1026	BWR	AL	5/11/1967	7/31/1974	12/31/1985	7.2	11.4	32.70%	800	966
Oconee 2	851	PWR	SC	11/7/1967	9/9/1974	9/9/2014	6.8	40	33.10%	611	729
Three Mile Island 1	790	PWR	PA	5/19/1968	9/2/1974	9/2/2014	6.3	40	30.60%	1579	1852
Zion 2	1001	PWR	IL	12/27/1968	11/14/1973	9/19/1996	4.9	22.8	32.50%	912	1025
Arkansas 1	836	PWR	AR	12/7/1968	12/19/1974	12/19/2014	6	40	30.80%	890	1036
Oconee 3	851	PWR	SC	11/7/1967	12/16/1974	12/16/2014	7.1	40	33.00%	611	735
Peach Bottom 3	1068	BWR	PA	2/1/1968	12/23/1974	12/23/2014	6.9	40	32.40%	1208	1443
Prairie Island 2	510	PWR	MN	6/26/1968	12/21/1974	12/21/2014	6.5	40	31.70%	1352	1596
Duane Arnold	535	BWR	IA	6/23/1970	1/30/1975	1/30/2015	4.6	40	31.30%	1677	1868
Browns Ferry 2	1087	BWR	AL	5/11/1967	3/4/1975	3/4/2015	7.8	40	32.70%	800	984

Name of Reactor	MW		State	Start of construction	End of construction	Shutdown	Years for construction	Lifetime	Thermal Efficiency	Overnight capital cost (2004 $)	Total capital cost (2004 $)
Rancho Seco	862	PWR	CA	10/12/1968	4/2/1975	6/7/1989	6.5	14.2	32.50%	1167	1377
Calvert Cliffs 1	834	PWR	MD	7/8/1969	5/8/1975	5/8/2015	5.8	40	31.40%	1450	1677
James A. Fitzpatrick	801	BWR	NY	5/21/1970	7/28/1975	7/28/2015	5.2	40	34.20%	1398	1585
Donald C. Cook 1	1013	PWR	MI	3/26/1969	8/28/1975	8/28/2015	6.4	40	31.80%	1318	1552
Brunswick 2	820	BWR	NC	2/8/1970	10/30/1975	10/30/2015	5.7	40	32.60%	1211	1396
Edwin I. Hatch 1	801	BWR	GA	10/1/1969	12/31/1975	12/31/2015	6.2	40	31.30%	1437	1684
Millstone 2	864	PWR	CT	12/12/1970	12/26/1975	12/26/2015	5	40	33.60%	1453	1640
Trojan	1085	PWR	OR	2/9/1971	11/21/1975	11/9/1992	4.8	17	32.50%	1233	1381
Indian Point 3	967	PWR	NY	8/14/1969	8/30/1976	8/30/2016	7	40	33.00%	1517	1822
Beaver Valley 1	814	PWR	PA	6/27/1970	10/1/1976	10/1/2016	6.3	40	31.40%	1774	2079
St Lucie 1	829	PWR	FL	7/2/1970	12/21/1976	12/21/2016	6.5	40	31.20%	1681	1983
Browns Ferry 3	1091	BWR	AL	8/1/1968	3/4/1977	3/4/2017	8.6	40	32.70%	800	1009
Brunswick 1	821	BWR	NC	2/8/1970	3/18/1977	3/18/2017	7.1	40	34.10%	1211	1457
Crystal River	823	PWR	FL	9/26/1968	3/13/1977	3/13/2017	8.5	40	32.00%	1384	1737
Calvert Cliffs 2	836	PWR	MD	7/8/1969	4/2/1977	4/2/2017	7.7	40	31.40%	1450	1780
Salem Creek 1	1106	PWR	NJ	9/25/1968	6/30/1977	6/30/2017	8.8	40	31.60%	1938	2456
Davis-Besse 1	877	PWR	OH	3/25/1971	7/31/1978	7/31/2018	7.4	40	33.60%	1890	2291
Joseph M. Farley 1	826	PWR	AL	8/17/1972	12/2/1977	12/2/2017	5.3	40	31.00%	1955	2224
North Anna 1	905	PWR	VA	2/20/1971	6/3/1978	6/3/2018	7.3	40	34.00%	1620	1960
Donald C. Cook 2	1063	PWR	MI	3/26/1969	7/2/1978	7/2/2018	9.3	40	31.90%	1318	1697
Three Mile Island 2	906	PWR	PA	11/5/1969	12/2/1978	3/28/1979	9.1	40	30.60%	1825	2336
Edwin I. Hatch 2	829	BWR	GA	12/28/1972	9/2/1979	9/2/2019	6.7	40	31.10%	1437	1706
Arkansas 2	858	PWR	AR	12/7/1972	3/26/1980	3/26/2020	7.3	40	29.80%	1437	1740
North Anna 2	908	PWR	VA	2/20/1971	12/14/1980	12/14/2020	9.8	40	34.00%	1620	2123
Joseph M. Farley 2	834	PWR	AL	8/17/1972	7/30/1981	7/30/2021	9	40	31.00%	1955	2492
Sequoyah 1	1137	PWR	TN	5/28/1970	7/2/1981	7/2/2021	11.1	40	33.50%	1488	2031
Salem Creek 2	1109	PWR	NJ	9/25/1968	10/13/1981	10/13/2021	13	40	31.90%	1938	2820

Name of Reactor	MW		State	Start of construction	End of construction	Shutdown	Years for construction	Lifetime	Thermal Efficiency	Overnight capital cost (2004 $)	Total capital cost (2004 $)
McGuire 1	1119	PWR	NC	2/23/1973	12/2/1981	12/2/2021	8.8	40	32.60%	1355	1718
Sequoyah 2	1126	PWR	TN	5/28/1970	6/3/1982	6/3/2022	12	40	33.50%	1488	2093
Susquehanna 1	1089	BWR	PA	11/4/1973	6/3/1983	6/3/2023	9.6	40	32.50%	2862	3722
San Onofre 2	1070	PWR	CA	10/19/1973	8/8/1983	8/8/2023	9.8	40	34.50%	2966	3885
St Lucie 2	836	PWR	FL	5/3/1977	8/8/1983	8/8/2023	6.3	40	30.70%	2704	3169
Lasalle 1	1079	BWR	IL	9/11/1973	1/1/1984	1/1/2024	10.3	40	32.20%	1954	2600
V C Summer	936	PWR	SC	3/22/1973	1/1/1984	1/1/2024	10.8	40	32.80%	2514	3398
McGuire 2	1114	PWR	NC	2/23/1973	3/2/1984	3/2/2024	11	40	33.30%	1355	1845
San Onofre 3	1080	PWR	CA	10/19/1973	4/2/1984	4/2/2024	10.5	40	34.50%	2966	3966
Lasalle 2	1080	BWR	IL	9/11/1973	10/20/1984	10/20/2024	11.1	40	31.80%	1954	2668
Columbia Generating St.	1103	BWR	WA	3/20/1973	12/14/1984	12/14/2024	11.7	40	33.90%	4589	6397
Callaway	1124	PWR	MO	4/17/1976	12/20/1984	12/20/2024	8.7	40	32.60%	3287	4154
Susquehanna 2	1105	BWR	PA	11/4/1973	2/13/1985	2/13/2025	11.3	40	32.60%	2862	3931
Diablo Canyon 1	1081	PWR	CA	4/24/1968	5/7/1985	5/7/2025	17	40	32.60%	3221	5351
Catawba 1	1130	PWR	SC	8/8/1975	6/29/1985	6/29/2025	9.9	40	33.80%	1906	2503
Grand Gulf	1186	BWR	MS	9/5/1974	7/1/1985	7/1/2025	10.8	40	30.60%	3473	4700
Wolf Creek	1158	BWR	KS	5/31/1977	9/3/1985	9/3/2025	8.3	40	35.00%	3168	3951
Byron 1	1137	PWR	IL	1/1/1976	9/16/1985	9/16/2025	9.7	40	32.80%	2595	3388
Waterford 3	1076	PWR	LA	11/15/1974	9/24/1985	9/24/2025	10.9	40	32.40%	3751	5082
Limerick 1	1110	BWR	PA	6/20/1974	2/2/1986	2/2/2026	11.6	40	32.10%	3778	5247
Palo Verde 1	1238	PWR	AZ	5/26/1976	1/28/1986	1/28/2026	9.7	40	32.10%	2777	3623
Diablo Canyon 2	1087	PWR	CA	12/10/1970	3/14/1986	3/14/2026	15.3	40	32.50%	3221	5041
Millstone 3	1138	PWR	CT	8/10/1974	4/24/1986	4/24/2026	11.7	40	34.20%	4279	5959
River Bend	953	BWR	LA	3/26/1977	6/17/1986	6/17/2026	9.2	40	29.90%	5602	7204
Catawba 2	1130	PWR	SC	8/8/1975	8/20/1986	8/20/2026	11	40	33.60%	1906	2596
Palo Verde 2	1281	PWR	AZ	5/26/1976	9/23/1986	9/23/2026	10.3	40	32.50%	2777	3700

Name of Reactor	MW		State	Start of construction	End of construction	Shutdown	Years for construction	Lifetime	Thermal Efficiency	Overnight capital cost (2004 $)	Total capital cost (2004 $)
Hope Creek	1044	BWR	NJ	11/5/1974	12/21/1986	12/21/2026	12.1	40	31.50%	5950	8400
Shearon Harris	886	PWR	NC	1/28/1978	5/3/1987	5/3/2027	9.3	40	33.70%	4747	6111
Vogtle 1	1145	PWR	GA	6/29/1974	6/2/1987	6/2/2027	12.9	40	31.40%	4162	6030
Byron 2	1120	PWR	IL	1/1/1976	8/22/1987	8/22/2027	11.6	40	33.50%	2595	3605
Beaver Valley 2	828	PWR	PA	5/4/1974	11/18/1987	11/18/2027	13.5	40	32.30%	6134	9069
Perry	1212	BWR	OH	5/4/1977	11/19/1987	11/19/2027	10.5	40	33.20%	4917	6595
Clinton	987	BWR	IL	2/25/1976	11/25/1987	11/25/2027	11.7	40	34.00%	6229	8686
Palo Verde 3	1241	PWR	AZ	5/26/1976	1/8/1988	1/8/2028	11.6	40	32.70%	2777	3857
Fermi	1077	BWR	MI	9/27/1972	1/23/1988	1/23/2028	15.3	40	33.80%	5265	8259
Nine Mile Point 2	1106	BWR	NY	6/25/1974	3/11/1988	3/11/2028	13.7	40	32.50%	6687	9942
Shoreham	820	BWR	NY	4/16/1973	4/21/1989	6/28/1989	16	40	32.50%	8165	13108
Braidwood 1	1144	PWR	IL	1/1/1976	7/29/1988	7/29/2028	12.6	40	33.10%	2549	3651
South Texas 1	1250	PWR	TX	12/23/1975	8/25/1988	8/25/2028	12.7	40	32.50%	4020	5777
Braidwood 2	1127	PWR	IL	1/1/1976	10/17/1988	10/17/2028	12.8	40	33.10%	2549	3678
Vogtle 2	1146	PWR	GA	6/29/1974	5/20/1989	5/20/2029	14.9	40	31.40%	4162	6435
South Texas 2	1250	PWR	TX	12/23/1975	6/19/1989	6/19/2029	13.5	40	33.60%	4020	5934
Limerick 2	1123	BWR	PA	6/20/1974	1/8/1990	1/8/2030	15.6	40	32.50%	3778	5972
Comanche Peak 1	1150	PWR	TX	12/20/1974	8/13/1990	8/13/2030	15.6	40	32.30%	7827	12412
Seabrook 1	1155	PWR	NH	7/7/1976	8/19/1990	8/19/2030	14.1	40	33.60%	4949	7457
Comanche Peak 2	1150	PWR	TX	12/20/1974	8/3/1993	8/3/2033	18.6	40	32.10%	4364	7653
Watts Bar 1	1121	PWR	TN	1/24/1973	5/27/1996	5/27/2036	23.3	40	32.50%	4604	9521

R1-6.2. The French Nuclear Construction Cost

References were collected on historical reactor costs in France for all the 58 French reactors, including overnight capital costs as made publicly available in early 2012 by the "Cour de Comptes", the French equivalent to the U.S. Government Accountability Office (GAO) (Rangel, 2012). The single, most complete previous estimate was made by Arnulf Grubler and was obtained indirectly by examining EDF financial data (Grubler, 2010).

Table R1-4 and Figure R1-5 show the construction cost of all 58 French power reactors currently in operations. They are divided in 3 "*palier*" or size categories: 900 MW_e (34 reactors), 1300 MW_e (20 reactors) and 1450 MW_e (4 reactors). In turn, the 900 MW_e *palier* is comprised of the CP0, CP1 and CP2 types; the 1300 MW_e *palier* is comprised of the P4 and P'4 type and the 1450 MW_e is of the N4 type. Types CP0, CP1, CP2 and P4 are Westinghouse-licensed designs (in blue in the figure), while the P4 and N4 types have been constructed allowing less standardization during construction (Rangel 2012), which may have led to a higher ultimate construction cost (in red in Figure R1-5). It is noted that the cost data in Table R1-4 had to be converted from the original un-escalated French Francs to Euros (which was done by the Cour de Comptes), and then to U.S. dollars (which was done using an exchange rate of 1.3, prevalent in 2013/2014). This sequence of conversions, by itself, is a source of uncertainty: therefore the cost numbers of Table R1-4 and Figure R1-5 should be considered only approximations.

Table R1-4. French nuclear plants' construction costs. Data source: Cour des Comptes, 2012 (Rangel, 2012).

Palier	Plant	MW	Criticality	Type	Cost (E2010/kW)	Cost ($/kW)
900 MW	Fessenheim1.2	1780	1978	CP0	836	1087
900 MW	Bugey2.3	1840	1979	CP0	886	1152
900 MW	Bugey4.5	1800	1979	CP0	899	1169
900 MW	Damprierre1.2	1800	1980	CP1	1,217	1582
900 MW	Gravelines1.2	1840	1980	CP1	822	1069
900 MW	Tricastin1.2	1840	1980	CP1	1,188	1544
900 MW	Blayais1.2	1830	1982	CP1	1,110	1443
900 MW	Dampierre3.4	1800	1981	CP1	1,172	1524
900 MW	Gravelines3.4	1840	1981	CP1	856	1113
900 MW	Tricastin3.4	1840	1981	CP1	1,247	1621
900 MW	Blayais3.4	1820	1983	CP1	890	1157
900 MW	Gravelines5.6	1820	1985	CP1	1,093	1421
900 MW	Saint Laurent 1,2	1760	1983	CP2	1,120	1456
900 MW	Chinon 1,2	1740	1984	CP2	1,148	1492
900 MW	Cruas1.2	1760	1984	CP2	1,119	1455
900 MW	Cruas3.4	1760	1984	CP2	1,253	1629
900 MW	Chinon3.4	1760	1987	CP2	978	1271
1300 MW	Paluel1.2	2580	1985	P4	1,531	1990
1300 MW	Paluel3.4	2580	1986	P4	1,157	1504
1300 MW	St Alban1.2	2600	1986	P4	1,129	1468
1300 MW	Flamanville1.2	2580	1987	P4	1,287	1673
1300 MW	Cattenom1.2	2565	1987	P'4	1,358	1765
1300 MW	Belleville1.2	2620	1988	P'4	1,083	1408
1300 MW	Cattenom3.4	2600	1991	P'4	1,149	1494
1300 MW	Nogent1.2	2620	1988	P'4	1,194	1552
1300 MW	Glofech1.2	2620	1992	P'4	1,305	1697
1300 MW	Penly1.2	2660	1991	P'4	1,227	1595
1450 MW	Chooz1.2	2910	2000	N4	1,635	2126
1450 MW	Civaux1.2	2945	2002	N4	1,251	1626

Figure R1-6 shows the construction times of the French nuclear plants (from Grubler 2010). While few of the plants had longer construction times (especially the later P'4 and the N4 types), the average construction time of 76 months is very short in comparison to the average US case of 108 months (Table R1-3). About half of the plants had construction times of less than 72 months (6 years).

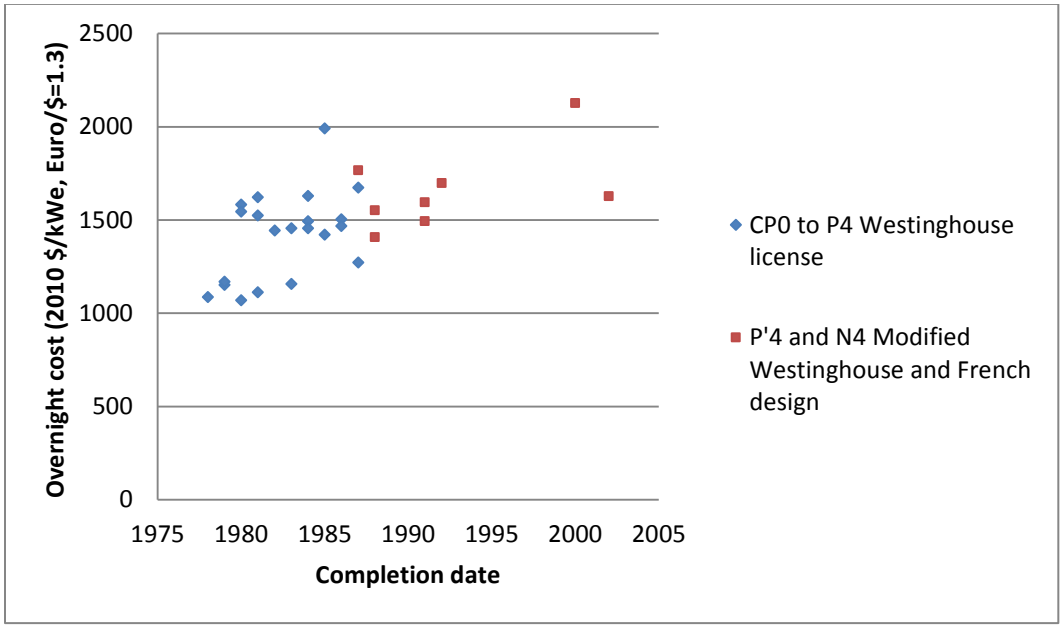

Figure R1-5. Construction cost for all 58 French reactors.

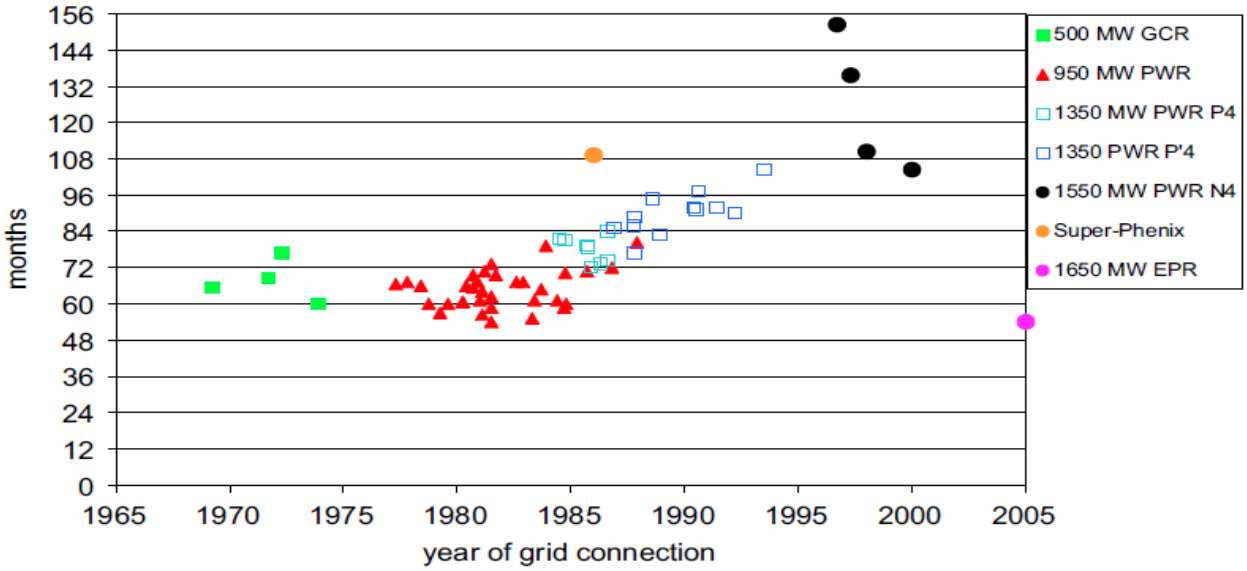

Figure R1-6. Construction times of the French plants (Flamanville unit 3, labelled "1650 MW EPR", is currently under construction).

It is of interest to understand the variation in costs associated with the French construction program, since it may lead to useful insight to quantify an appropriate uncertainty distribution for the "engineering" construction costs. A useful metric for that purpose is the ratio of the "standard deviation" to the average cost. Table R1-5 shows that for the entire construction program, the ratio of "standard deviation/average" was 17%. However, within the construction period, there was a clear increase in cost, as the construction

program evolved (Figure R1-5): the standard deviation of the entire series would also "capture" the up-trend in cost, which is not relevant to intrinsic uncertainty of the "engineering" or un-avoidable costs. For this reason, the "de-trended" data is also presented in the last column of Table R1-5.

First the trend-line has been calculated, leading to the following Equation:

$$Cost = 26.042 \cdot Year - 50216$$

Afterwards, the trend values have been subtracted from the actual construction cost to obtain the "de-trended" data.

As expected, the average of the "de-trended" line is now close to zero, and MAX and MIN represented then maximum and minimum deviation from the trend line. The standard deviation has also been reduced from 251 $/kW$_e$ to 201 $/kW$_e$, thus reducing the ratio of "standard deviation/average" from 17% to 13%. Further subdivisions can be made by power level (i.e. the "*palier*" type) and by the level of design standardization: "*Palier* 1300" had the lowest standard deviation/average construction cost ratio, at 11%, while the highest was for "*Palier* 1450", at 19%; however the "*Palier* 1450" values are based on only two data points. The two large groups "largely Westinghouse-based" (i.e. the CP0 to P4) and the "largely French design" P'4 and N4 had ratio of "standard deviation/average" construction cost of, respectively, 17% and 13%.

Table R1-5. Min, max, average and standard deviation of reactors constructed in France 1978-2002 (all in $/kW$_e$ with a €/$ exchange rate of 1.3).

	All reactors	Palier 900	Palier 1300	Palier 1450	CP0 to P4	P'4 and N4	All reactors (de-trended)
MAX	2125.5	1628.9	1990.3	2125.5	1990.3	2125.5	512.9
MIN	1068.6	1068.6	1407.9	1626.3	1068.6	1407.9	-293.8
AVERAGE	1485.6	1363.8	1614.6	1875.9	1420.0	1657.8	0.1
stdev	251.0	201.0	173.0	353.0	234.4	219.5	201.2
stdev/average	*17%*	*15%*	*11%*	*19%*	*17%*	*13%*	*13%*

R1-6.3 Best Experience in Reactor Construction (i.e. "Engineering Construction Cost")

The objective of this section is to draw a conceptual distinction between (1) construction cases in which cost overruns were minimized, i.e. the "best experiences" in reactor constructions and (2) cases that experienced substantial cost overruns; thus establishing a framework for understanding the reasons for the observed reactor capital costs, including the fundamental drivers of costs and the main reasons for the biggest cost overruns observed historically.

Most of the literature that dwells on reactor construction's historical costs attempts, through econometric techniques, to interpolate and extrapolate from historical data, often coming to the conclusions that future reactor costs can have very large and very uncertain costs. (See for example, (Koomey and Hultman 2007, Komanoff 1981, Rangel and Leveque 2012).

What is done here is, instead, different. The main focus will be on those constructions in which the process went reasonably well from a cost perspective, thus establishing a basis for costs in the "*best experience*" cases. The numerous cases in which costs escalated, sometimes dramatically, offer the opportunity to understand what underlying issues are most often responsible for the cost escalation.

A key driver of cost overruns during construction was found to be *the degree of changes requested during the construction phase* of a new reactor. These changes are, in turn, typically driven by (1) incomplete engineering at the start of construction (i.e. *engineering instability*); and (2) requests for alteration induced by regulatory changes after the construction has started (i.e. *regulatory turbulence*).

If the design is fully completed before the construction starts and no changes are required during the construction phase, complex construction projects can be kept reasonably within budget by the standard use of "*fixed price*" contracts, negotiated with competitive bidding: the efficiencies built into this process minimize the construction costs. However, if design changes significantly during the construction phase, the original "*fixed price*" contracts become un-tenable, since re-bidding is usually impractical. Therefore, in these cases it becomes necessary to switch the "*fixed-price*" contracts to "*cost-plus*" contracts, and efficiencies are lost, generating potentially substantial cost overruns, while the utility and architect-engineer lose control over the contractors' expenditures. On the other hand, fixed price contract arrangements *"create incentives to work efficiently and expeditiously to maximize the contractor's profit"*; while *"the objective under cost-plus contracts is to maximize total revenues, which shifts the prevailing orientation towards longer schedules and greater expenditures."* (Komanoff 1981).

The following are among the main causes of cost-increases due to design alterations during the construction phase:

- Completed work need to be removed and/or altered, with the obvious cost of removal (which can be substantial, such as for example for reinforced concrete that has already cured) and reconstruction. An important cost effect of reconstructions is that it often affects nearby systems as well (an effect known as "systemic effect", typical of nuclear power plants), which otherwise would not have been directly affected by the changes, either (1) because of logistical constraints (e.g. having to remove and later reinstall piping already installed in order to insert a new piece of equipment that was not in the original design), or (2) because of errors (e.g. inadvertently damaging equipment already in place when adding a new piece of equipment), or (3) because contractors previously released from the site had to be re-called and retrained. For example, the following is a quote from a testimony from PG&E before the California Public Utilities Commission: *"There are significant inefficiencies in trying to design to fit existing buildings and installed components. […] Work has to be done out of sequence in a restricted access and work area."* (Brand 1979).
- Construction sequences have to be altered, and equipment delivery schedules have to be altered, potentially idling groups of workers while waiting for the new equipment or for the re-design to be completed. This can dramatically reduce the labor productivity and increase the labor costs.
- Increased construction duration can produce a self-reinforcing feedback loop, more important during periods of high regulatory turbulence, by exposing the project to the risk of additional regulatory changes. Additionally, longer construction duration increases directly the "Interest during construction" costs and can further disrupt the construction logistics.

Stricter regulatory oversight, which generally increased with increasing regulatory turbulence throughout the 1970s and 1980s, also increased the costs overruns (1) by increasing the direct labor costs through the extra requirements for supervision and compliance; and (2) by largely impeding the "dynamic" engineering adaptation and "on-the-spot" problem solving that may have otherwise easily corrected minor challenges that arose during the construction phase. The procedures required, instead, to re-submit the modified design for a new regulatory approval, which cost delays that resulted in idled skilled crews and un-expected capital charges. (Kessides 2012).

Additionally, there is a self-reinforcing feedback loop between scope changes and regulatory oversight. In an environment of frequent changes to the on-going work that require re-construction, it is not un-common for staff to start developing negative expectations on the outcome of work, which can, by itself, reduce productivity. Additionally, when workers start anticipating that a job will be redone several times, because of changes to regulations and/or design alterations, they may be less careful during the installation, which can lead to high rejection rate by the regulators, completing the feedback loop between scope changes and regulatory oversight (Komanoff 1981).

It was also found that the capital cost escalation of nuclear power plants – already observable for plants completed in the pre-TMI period (i.e. pre-1979) – has been primarily driven by increasing

regulatory stringency, which in turn has been quantitatively associated with the overall expansion of the nuclear power sector. The increased regulatory stringency manifested itself in: (1) the application of more stringent and explicit safety standards, which caused a direct increase in the amount of labor, material and equipment required to build nuclear plants; and (2) the expansion of the regulatory effort, requiring greater documentation and standardization of regulatory requirements: this mostly caused a substantial increase in labor costs. Both effects caused a fundamental increase in the cost of nuclear plants. None of the several studies attempted in the past, however, could establish a quantitative relational link between increased requirements for commodities and labor driven by specific regulatory changes, and the ultimate cost of new construction. This is because nuclear power plants feature substantial "systemic effects", for which the cost of altering/adding equipment propagates beyond the system being altered, and is in contrast to the case of coal, for example, for which most cost increases in the 1970s could be quantitatively traced to the addition of individual components mandated for pollution control, the effects of which on the plant construction cost could be isolated.

R1-6.5. Reconciliation to the Late 1980s Cost Observations with the Currently Observed Construction Costs

The base construction costs of the EEDB "Better Experience" PWR was observed to be 1272 $\$/kW_e$ in 1987 dollars (ORNL 1988b), which would be 2658 $\$/kW_e$ in 2014 dollars using the Consumer Price Index (CPI) general inflation index between 1987 and 2014. This amount does not include owner's costs and contingencies, which typically add about 10% each to the base construction cost, to arrive at the overnight cost. After adding owner's costs and contingencies, the resulting observed overnight cost would be about 3200 $\$/kW_e$.

However, it was also calculated in (ORNL 1988b) that between 1978 and 1987, the total construction costs for the "Better Experience" PWR increased by 3% points annually *in real terms* (i.e. above the rate of general inflation), while the construction costs for the "Median Experience" PWR increased by 10% annually above the rate of the general inflation.

Since the EEDB report series had no further updates after 1988, but construction continued in the United States until 1996 (when the first criticality of Watts Bar 1 was reached), it is of interest to calculate the expected current Best and Median Estimates construction costs if: those cost increases had continued at the same rates while construction was on-going in the U.S., and then just increased with the general rate of inflation after the construction program stopped. The logic for this is that construction cost increases above the rate of inflation were found to be mostly driven by increased regulatory stringency (Ganda 2014), which in turn was found to be strongly correlated with the overall expansion of the nuclear sector (Komanoff 1981). Based on these considerations, it can be argued that no real cost increase is to be expected during periods of non-expansion in the overall nuclear sector, such as the period between 1996 and 2014 for the U.S. Using this approach, the overnight construction cost of (EEDB 1988) for the best experience cases would have been escalated to 4181 $\$_{2014}/kW_e$, which is similar with the pre-construction estimates of Vogtle and VC Summer (Ganda 2014). Additionally, this cost is roughly consistent with the 4 identical reactors being built in the UAE, United Arab Emirates, by South Korea, having a construction cost of about $3650/kW. These plants are replicas of two reactors currently being built in South Korea. If a project is an exact replica of a previous identical construction, it is conceivable that *the home office engineering services* cost (of about 505 $/kWe) could be mostly avoided: these costs are primarily associated with the engineering and design, project planning and associated overhead. However, if a project is a replica of a previously well-executed project, this cost should be minimal, since little new engineering should be necessary, and any work by the home office service should be just incidental. Eliminating the 505 $/kWe cost of the *home office engineering services* cost from an expected cost of 4100 $/kWe leads to a cost very similar to the outcome of the UAE project, of about $3650/kWe.

R1-6.6. The Probability Density Function That Best Approximates the U.S. Historical Data Distribution (i.e. the Non-Engineering Costs)

Historical U.S. construction cost data can be used to gain a quantitative understanding of the "non-engineering" construction cost, since they include a wide range of actual outcome, several of which included substantial cost overruns. The key assumption when using this data would be that, in the future, many of the mistakes done in the past could be repeated, leading to similar quantitative outcomes.

In general, the use of raw historical construction cost data is *not* recommended for fuel cycle equilibrium economic analyses, such as those performed for the Fuel Cycle Evaluation and Screening in 2013 (Ganda et al. 2013**),** for which an "engineering cost" probability density function is instead discussed in Section R1-6. The assumption is that, having reached a status of equilibrium, the construction capabilities, including the regulatory infrastructure, are sufficiently mature as to avoid the typical causes of cost overruns observed in the past. However, it still can be useful to have the capability to use "non-engineering" probability density functions for other types of economic evaluations based on historical data. It is also useful to test the validity of functional forms for uncertainty distributions of nuclear plant costs that should be grounded in historical data.

For the purpose of performing Monte Carlo calculations of the LCOE using historical construction data, it is of interest to identify the probability density function that approximates best a bar plot of such data.

It was found that a good approximation could be obtained by a log-normal probability density function, which appears better fitting than other functional forms of the distributions: Figure R1-8 shows a bar plot of the historical overnight costs of U.S. reactors (in 2004 $/kWe), and a log-normal probability density function that approximate the historical curve. The parameters of the curve, together with the equation used to generate them based on the historical data, are listed below:

- Probability density function (pdf): $p(x) = \frac{1}{x\sqrt{2\pi\sigma}} e^{-\frac{(\ln x-\mu)^2}{2\sigma^2}}$.
- Median of the distribution: 1825 $\$_{2004}/kW_e = e^{\mu}$, therefore μ=7.5.
- Mean of the distribution: 2450 $\$_{2004}/kW_e = e^{\mu+\frac{\sigma^2}{2}}$, therefore σ=0.767.
- Standard Deviation of the distribution: $std = \sqrt{\left(e^{\sigma^2}-1\right)e^{2\mu+\sigma^2}}$=2171 $\$_{2004}/kW_e$.
- Standard Deviation/Mean=2171/2450=88.6%.

Additionally, the log-normal probability distributions should be truncated to best represent the historical U.S. construction data. The truncation can be implemented, for example, through the method of "rejection sampling" if a Monte Carlo approach is used for the simulation, as is currently implemented in the NE-COST code. A resulting sampled probability density function is shown in Figure R1-9.

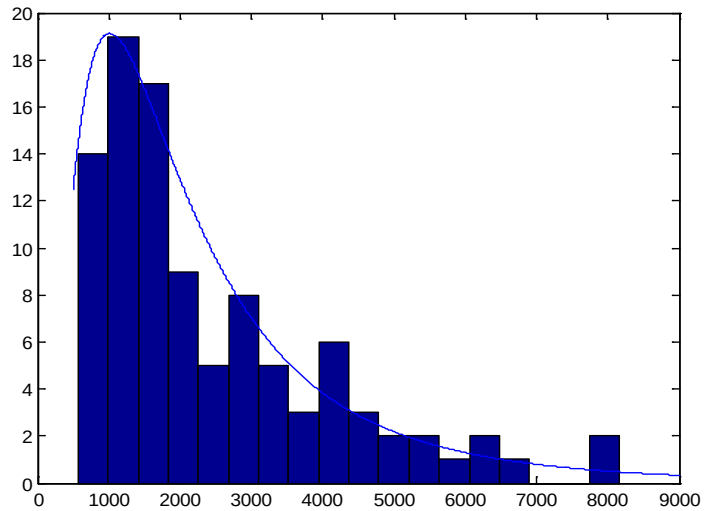

Figure R1-8. Historical overnight costs of U.S. reactors (in 2004 $/kWe) and fitting log-normal pdf.

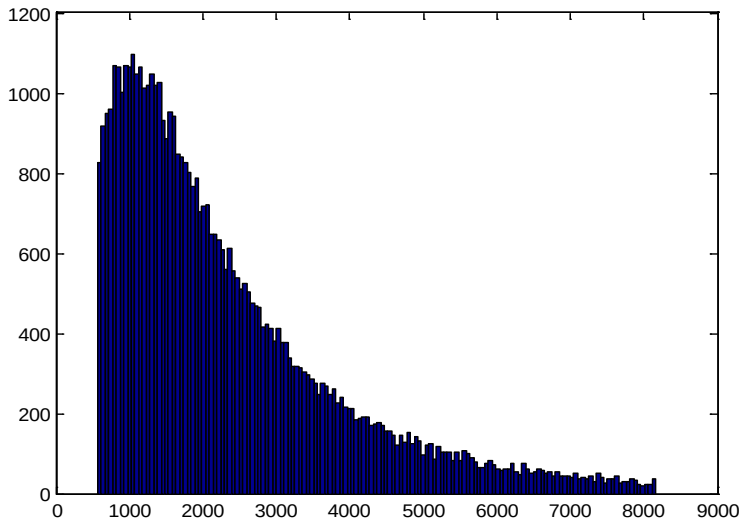

Figure R1-9. Truncated Log-normal probability density function of the historical US reactor costs (in 2004$) as sampled by a Monte Carlo sampler using the previous equations.

R1-6. Summaries of Engineering and Non-Engineering Costs, and Suggested Uncertainty Distributions

Engineering costs: expected values and uncertainties distributions.

It was found that the current expected value of the "engineering" construction cost of LWRs in the U.S. is $4100/kWe, in 2015 dollars, based on the pre-construction estimates of the current construction projects, as well as on the observations made in the late 1980s and reconciled to today's dollars using the methodology described in section R1-6.5. This value is also consistent with the outcome of the well-executed UAE construction project, as explained in Section R1-6.5

It was also found that the historical standard deviation for the "engineering" construction cost of U.S. LWR is about 30% of the expected "engineering cost", when using the pre-construction estimates as approximations for "engineering", or avoidable costs. It was also found that the reactors constructed in France had substantially lower cost standard deviations (as a ratio of expected values) than the U.S. pre-construction estimates (of between 11% and 19%, depending on the observation subset), even though the French data are based on the actual construction outcomes, and the U.S. data are based on the budget, or

pre-construction estimates. The reasons for this difference could be attributable to a higher construction standardization in France, and to the high rates of cost overruns in the U.S., which may have led the estimators to insert a higher (and varying from project to project) contingency allowance in the preconstruction estimates. However, further investigation should be performed in this area to better understand the reasons for the observed differences.

Adopting as reference the de-trended data, it is recommended to use a ratio of standard deviation to mean value of 13%, or 15% for a greater level of conservativeness, for the "engineering" construction cost uncertainties. Additionally, it is recommended to use a lognormal distribution to describe quantitatively the uncertainty associated with the contingency.

Therefore, the following are the parameters of the log-normal LWR overnight cost, plotted in Figure R1-10:

E= 4100 $\$_{2014}/kW_e$ and STD=600 $\$_{2014}/kW_e$, (i.e. 15% of E), then $\mu=8.2829$ and $\sigma=0.14917$.

Non-Engineering costs: expected values and uncertainties distributions.

Regarding the "non-engineering" expected costs and uncertainties, i.e. for the cost that could include overruns, construction delays etc…, it is recommended to use the historical U.S. data (in Section R1-6.6.) as a guide. This data can be used when the assumption is made that future mistakes may lead to a similar range of outcomes as those of the past. However, the historical data do also include early constructions, for which the cost was substantially lower than what could be observed currently in the U.S. (the lower cost can be mostly attributed to substantially less stringent regulatory requirements: please see (Ganda 2014)). Those early lower construction cost lowers the mean of the historical data to a value that is substantially lower than what appears achievable today, even under the best circumstances. For this reason, it may be advisable for the reader to use the historical construction cost data information (1) to obtain the functional for of the historical uncertainties, which is well represented by a log-normal pdf; and (2) to obtain information on the standard deviation of the historical construction costs *as a fraction of the historical mean*, which was about 90% of the expected value, in order to calculate the standard deviation of the currently expected construction cost.

With this approach, the user could chose a lognormal probability density function with an expected value of 4100 $/kWe and a standard deviation of 3700 $/kWe to model the "non-engineering" costs. Since it cannot be ascertained with full confidence that this distribution should be truncated at a particular value (as was instead the case for the actual cost data), it can be sampled as un-truncated, at the discretion of the user.

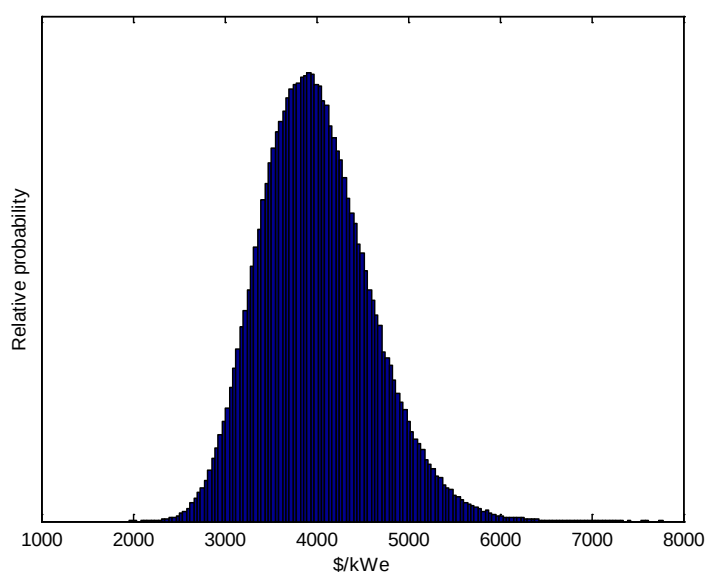

Figure R1-10. PDF of the LWR overnight cost with an expected value of 4100 $\$/kW_e$, a standard deviation of 15% of the expected value and a log-normal functional form.

R1-6. Capital and O&M Cost Distributions

If the log-normal cost distribution is not desirable and a more simplified triangle distribution is preferred as is common throughout the AFC CBR, the suggested distribution based on the analysis and distribution discussed prior for overnight capital cost is provided. The fixed and variable operation and maintenance were escalated from the AFC CBR 2015 with an escalation factor of 1.032 and rounded. Figure R1-11 shows the triangular distributions for overnight specific cost and the two components of the O&M cost for LWRs. The means (expected values) for each distribution are also shown on the graphs. These mean values should be used when discussing "single point" estimates without reference to uncertainty. All values are in year 2017 dollars.

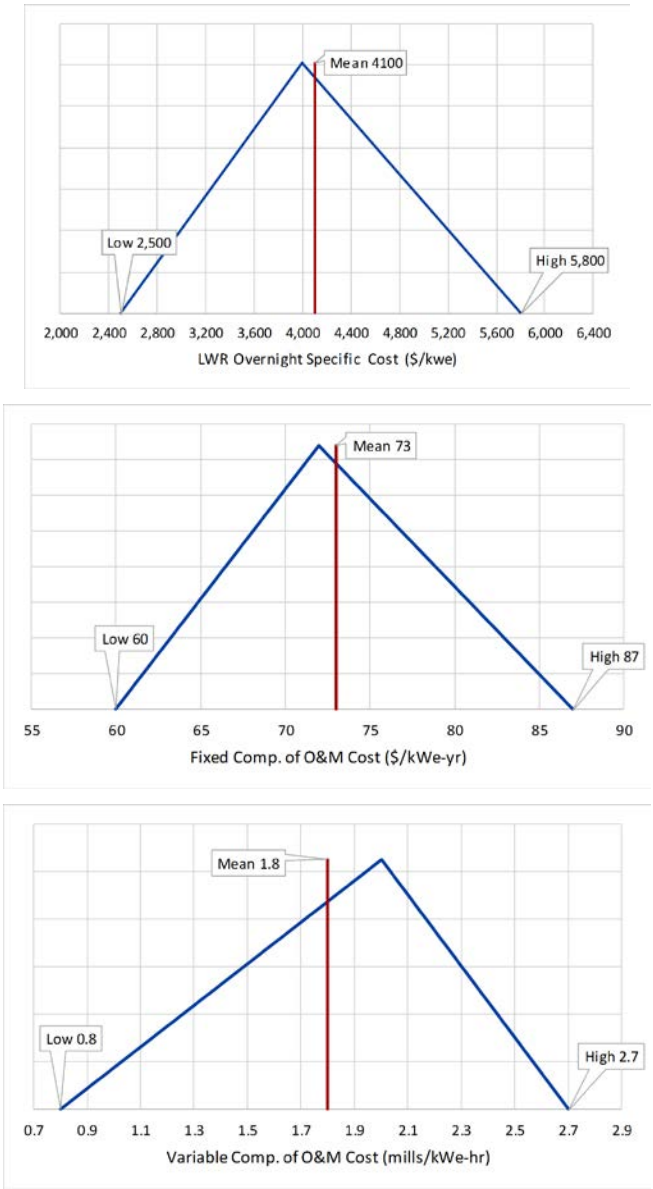

Figure R1-11. Probability Distributions and Means for LWR Cost Parameters.

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Module R2

Fast Reactors

Module R2

Fast Reactors

R2.MD SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2012 AFC-CBR)
- **Estimating Methodology for latest (2012 AFC-CBR) technical update from which this 2017 update was escalated:** The methodology described in Section RP2 of the earlier AFC-CBRs (now a supplementary document in the 2017 version entitled: *A Proposed Methodology for Transformation of Reactor Cost Data to the "What It-Takes" Table*, See CBR SD6) was applied to the available cost and design information for the with sodium fast reactors(SFRs) to extrapolate this available information to estimate the implied cost for an Nth-of-a-kind, well-built, well-designed, and well-executed fast reactor project at optimum commercial scale. The historical and very high $/kWe values resulting for many very small FR demonstration reactors when projected to large commercial scale FR projects often suggest costs ranges that are very high. When the above methodology is applied to real LWR and SFR design data these SFR projected costs turn out to be more comparable to LWRs and not to the sky high SFR costs often quoted as a result of mere extrapolation. The projected range of historical costs for the different projects; however, was considered and was then combined with engineering judgment to give the range and distribution for the 2012 AFC-CBR.

R2.RH REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2004 as Module R1.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2012
- **New technical/cost data which has recently become available and will benefit next revision:**
  - New methodology being considered for estimating costs of new advanced reactor concepts that looks at scaling based on different features (e.g., whether it scales based on thermal or electric power). Also there is significant interest in lead-cooled fast reactors and molten-salt fueled fast reactors and all current data is for sodium-cooled fast reactors which my require addition of new Modules or sub-modules for different fast reactor types.

R2-1. BASIC INFORMATION

The reactor is the central facility of the overall energy system and is supported by the overall fuel cycle. This section deals with "fast" reactors, which are those reactors in which the average neutron energies are in the higher energy or "fast" range (>0.1 MeV) for which less or no moderation is required. This allows the use of coolants that are higher in atomic number, including liquid metals such as sodium or lead, or even liquid salts. There are at present no operating commercial reactors in the U.S. of this type. However, small units, such as Fermi-I and Experimental Breeder Reactor (EBR)-II, produced power in the past. A large demonstration project, the Clinch River Breeder Reactor Project (CRBR), existed as a project in the 1970s and 1980s, but never got much beyond the design stage, and was terminated in 1983. Construction work on the CRBR had begun and some large equipment had been procured and fabricated when the project was canceled. The largest projects have been built in Russia (BN-600), and France (Superphenix). Russia, India, and China are the only countries presently constructing new fast reactors,

the BN-800 (Russia), the 500 MWe Prototype Fast Breeder Reactor (India), and the Chinese Experimental Fast Reactor (CEFR). As with thermal reactors, the predominant product from fast reactors is electricity. However, the heat that is generated can also be used for industrial applications such as hydrogen production (lower-temperature hydrogen production processes), district heat, process heat, or water desalination.

Fast reactors have the advantage over light-water reactors (LWRs) in that the fast spectrum provides better (lower) parasitic absorption to fission ratios resulting in more efficiently burning of the fissile isotopes along with additional surplus neutrons that can be used to transmute some fission products, consume transuranics from thermal reactor recycled spent fuel, and/or breed additional fissile material. The thermal spectrum reactors (e.g. LWRs) require additional fissile support, generally in the form of enriched uranium, and can therefore only consume a small fraction (on the order of 1%) of the initial uranium ore. Additionally and possibly more importantly, the fast spectrum reactor performance is far less sensitive to isotopic variation in the fuel composition, which will vary widely, depending on the source and age of the feed material being recycled.

Closing the fuel cycle is a significant part of the mission projected for the fast neutron reactor. In this case, a waste management mission (transmutation) can be accomplished in addition to electricity production. Fast reactors can also be used to convert fertile U-238 to fissile Pu-239 and Th-232 to fissile U-233, which makes for a highly-sustainable fuel cycle. This concept is known as “breeding,” and the reactors are known as fast breeder reactors. A fast-neutron nuclear power plant may actually consist of more than one “unit” or reactor on the same site. In fact, there are several concepts for modular sodium-cooled fast reactors that could be located in a reactor park along with dedicated fuel cycle facilities for integrated spent fuel recycle and refabrication.

The fuel cycle cost for a fast reactor (FR) is just one of the main four components of the busbar levelized unit electricity cost. (“Busbar” cost refers to the fact that the cost of electricity is that at the plant electrical boundary connection [busbar] and does not include distribution or other utility overhead costs.) As in Module R-1, the four components of the levelized unit electricity cost are:

1. Capital component (recovery of total project capital plus financing costs).
2. Operations and maintenance (O&M) component (annual nonfuel costs including manpower). Refueling manpower is usually carried in this major account.
3. Fuel cycle component (the sum of the relevant costs for the needed fuel cycle steps [modules] converted to mills/kWh or $/MWh unit costs). For the transmutation fast reactor fuel cycle, this account would include the pre-FR irradiation costs of processing the actinide products received from an LWR reprocessing facility, which will then serve (at minimum) as the startup fuel for the fast reactors.
4. Decontamination and decommissioning (D&D) costs, a fund accumulated to cover D&D of the reactor at its end-of-life.

Of these costs, the capital component for the fast reactor will always be the largest (as is the case for thermal reactors). This is different than other electricity generation sources, such as oil, natural gas, or coal, where fuel costs can be predominant and also unstable. The low fuel cycle cost is one of the advantages of nuclear power and is due to the fact that nuclear fuel delivers nearly one-million times the energy per unit mass compared to chemical fuel sources such as fossil fuels. The high capital cost of nuclear power is in part because of the need to include safety features (e.g. containment building) to confine radioactive materials originating in the reactor core during accidents. With fast reactors, there is also the fact that the main coolant candidate is liquid sodium, a reactive metal that will burn in air or

when contacted by water. Nuclear power plants are built to safety and quality control standards that exceed in breadth and scope that of fossil-fueled power plants.

The most useful cost figure of merit here is the specific total overnight construction cost, which is the cost of planning, designing, licensing, constructing, and starting up the reactor (up-front costs) divided by the net power capacity. It is usually expressed in $/kilowatt electric or $/kWe. One must be careful to specify whether the capital cost includes financing (interest) and other owner's costs. If the financing (interest) is excluded from the capital cost, this cost figure is called the "overnight" capital cost and is the best measure to compare costs from plant to plant. The total capital cost (TCC) includes interest during construction, which can be a significant percentage of the overnight cost if project construction or regulatory delays are encountered. The discussion below will deal mostly with the "overnight" expression of the specific capital cost, because it is most dependent on the reactor technology and also the one which appears most frequently in the literature.

R2-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

Fast reactors have a higher minimum fissile content to achieve criticality than thermal reactors which with low absorbing moderators like deuterium and carbon can operate with natural uranium, while fast reactors require fissile contents over and possibly well over 10% depending on the design and fuel composition. Most international (outside the United States) experience is with MOX or mixed oxide (urania, plutonia) ceramic fuel clad in stainless steel rods (Module D1-4). The fissile content for the MOX driver fuel is generally 17% or more of the heavy metal mass. The rods, typically thinner than those for thermal reactors, are bundled into fuel assemblies that are inserted into the reactor prior to startup. The United States has extensive experience with metal-based fast reactor fuels (Module D1-6) in EBR-II and Fast Flux Test Facility (testing only). The advantage of metal fuel is heat removal (high thermal conductivity) capability, compatibility with sodium coolant, passive-safety response characteristics during beyond design basis accidents, high breeding capacity, ease in fabricability, and its compatibility with electrochemical spent fuel recycling schemes. The internal heat generated by fission in the fuel is removed by the flowing liquid metal coolant and transferred by heat exchangers to steam generators where water is turned into steam. The steam then flows to turbine generators where electricity is generated. Because thermodynamic cycles are involved, most of the heat energy is rejected to the environment, as is true of all power plants using fossil or nuclear fuel. The ratio of the electric power generated to the total heat generation is the thermodynamic efficiency. Because of the higher liquid sodium temperature, the fast reactor is thermodynamically more efficient than the LWR.

Other reactor performance measures are the capacity factor and the fuel burnup. These have the same definitions as those for thermal reactors in Module R1.

R2-3. PICTURES, DIAGRAMS, AND DEPLOYMENT STATUS

Figure R2-1 shows the flow concepts within a pool-type fast reactor using a liquid sodium coolant. The other common configuration is a loop-type which is very similar in configuration to a PWR except that pumps and heat exchangers are located outside the primary reactor vessel (in the pool configuration, all primary system components are inside one [larger] reactor vessel). In either case there is a secondary sodium loop or loops that contain non-radioactive sodium. These "IHXs" isolate the steam system from radioactive sodium in the primary vessel.

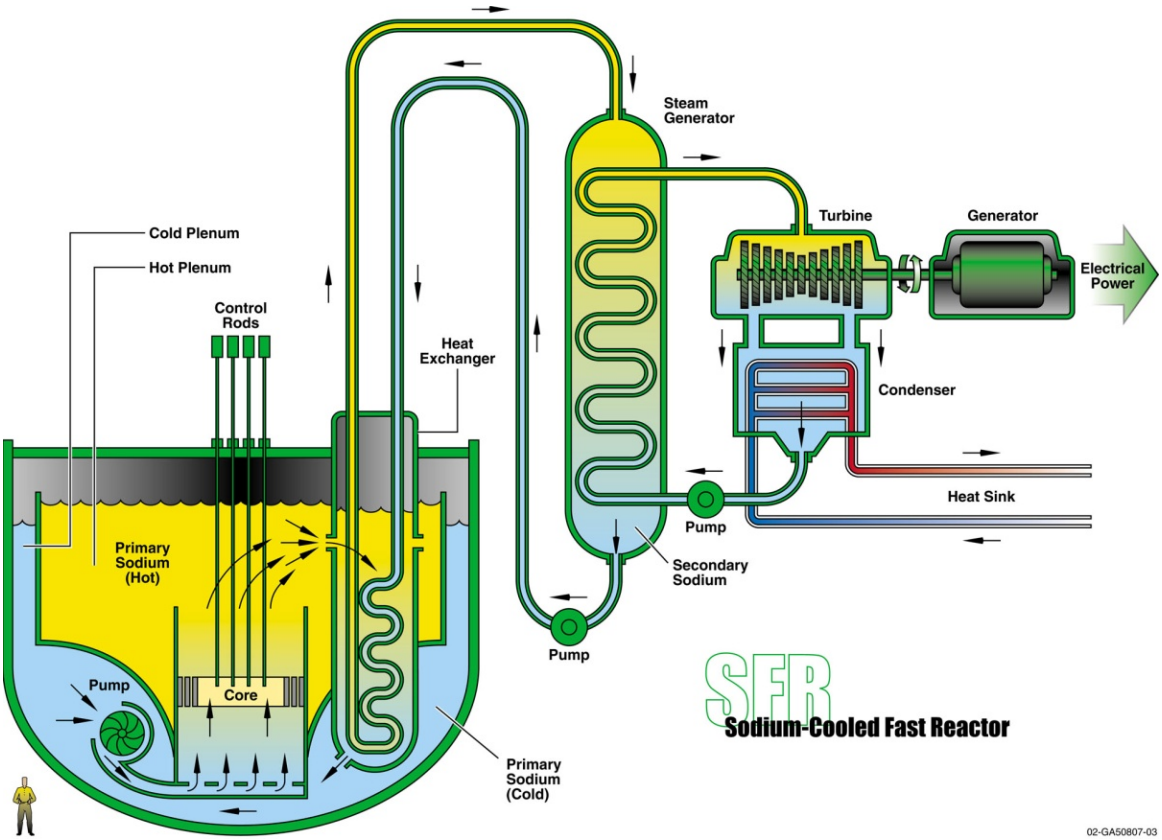

Figure R2-1. Major elements of a liquid-metal cooled fast reactor.

The last fast reactor project actually completed in the U.S. was the Fast Flux Test Facility (FFTF) at Hanford, Washington. This now-defueled and drained Department of Energy (DOE)-owned reactor was not designed to produce electricity; however, the addition of a power generating balance of plant was considered at one time.

R2-4. MODULE INTERFACES

The reactor will receive fuel assemblies from the fuel fabrication plant for its startup fast reactor fuel. Initially, this startup fuel is expected to be a more conventional fast reactor fuel, such as U-Pu-Zr metal fuel, mixed oxide fuel, or low to medium enriched uranium metal fuel, as appropriate. As transuranic-based fuels are fabricated and qualified for reactor use, they will be converted over to the transuranic-based fuel.

After irradiation, fuel assemblies are typically placed in in-vessel storage for one cycle prior to transfer, and then stored in a special area in liquid sodium until they decay to a degree that handling can be accomplished in air. Once the fuel assemblies have decayed sufficiently to be cooled passively by air, the fuel assemblies might be moved to storage casks for onsite or offsite storage to await processing. Direct transfer to a reprocessing facility (Modules F1 or F2/D2, depending on whether aqueous or electrochemical fuel reprocessing is used) is also possible. With integral fuel cycles, such as the Argonne National Laboratory (ANL) electrochemical recycle scheme based on metal fuel, reprocessing and new fuel fabrication take place in the same hot-cell facility, thus eliminating the need to transport spent FR fuel offsite which significantly reduces the external cycle time. For high growth scenarios, doubling time (the time to double the fissile inventory necessary to operate the fast reactors) is limiting and is a strong function of the external cycle time since the inventory required to operate a fast reactor includes all the

fissile material in the reactor plus all fissile material in storage, transport, process, and fabrication destined to be return to the fast reactor to complete the cycle. If the fuel residence time is 5 years, and the total external cycle time is 10 years (not unreasonable for off-site), the total fissile requirement is 3 times the reactor inventory. For on-site, the external cycle time of 2.5 years, would reduce the total required fissile inventory by half which would reduce the doubling time by a factor of 2 for identical reactors. Besides doubling time there may be other potentially desirable benefits such as proliferation benefits of lower out of core inventories and elimination of transport of the intact fast reactor fuel with its very high radiation levels, high fissile content, etc.

R2-5. SCALING CONSIDERATIONS

In general, the cost of electricity and the specific capital cost decrease with higher reactor size (electrical generation capacity). There is likely to be a point where factory production of small reactor modules, as opposed to traditional onsite construction, will allow reduction of unit costs. Studies, such as those being pursued by the Generation IV Economic Modeling Working Group, are investigating this issue. In earlier AFC-CBRs, Section R4 (Modular Reactors) addressed some of the issues of small and medium-sized reactors. It should be noted that the General Electric PRISM FR (aka ALMR or Advanced Liquid Metal Reactor) concept is modular in nature and is the subject of several papers and reports (Ehrman and Boardman 1995; Dubberly et al. 2003a; Dubberly 2003b; Fletcher 2006, GE Hitachi 2008).

R2-6. COST BASES, ASSUMPTIONS, AND DATA SOURCES

The question often arises as to how the specific overnight cost of a fast reactor compares to that for LWRs. From a pure engineering standpoint, one would expect pipe sizes to be larger because of the lower volumetric heat capacity of liquid sodium vis-à-vis liquid water (Forsberg 2007). There are also the additional safety and material considerations associated with the use of liquid sodium, a chemically-reactive metal. The core size, however, for a fast reactor can be smaller, since higher fissile content means less heavy metal content compared to LWRs. Other recent studies (Hoffman 2004) have looked at the issue of capital cost as a function of the conversion ratio. The well-defined Power Reactor Inherently Safe Module (PRISM) reactor was used as the basis for this study. There seems to be a feeling among some utilities and other stakeholders that the specific capital cost for an nth-of–a-kind (NOAK) fast reactor will be 1.0 to 1.6 times that for NOAK LWRs. There are, however, no recent detailed studies to backup this claim. There are several nations pursuing prototype fast reactors (Williams 2009); however, cost information is sketchy, and the prototypes are not the size of the eventual commercial unit that might be deployed. It is known that the Japanese and French are carefully considering the “lessons learned” from past FR projects to reduce this “FR/LWR” specific capital cost factor for their new concepts such as the Japanese Sodium-cooled Fast Reactor (JSFR), the European Fast Reactor (EFR), and the prototypes (Mainichi Daily News 2006; Platts 2006) that will precede them. However, as discussed above, it should be noted that fast reactors are viewed by many as the ultimate solution for closing the nuclear fuel cycle and have capabilities in regards to transmutation and sustainability that cannot be accomplished with LWR technology.

It is now useful to consider capital costs for fast reactors actually built or proposed (paper studies only for the latter). Cost and capacity information sometimes appear in trade press and general press sources. Utilities and architect engineers do not typically publish costs for their projects, especially under today’s environment of less economic regulation. Table R2-1 shows some historical data for FR projects actually completed and projected data for a few that have been recently announced. Tables R2-2 and R2-3 shows similar, but more detailed, data for FR projects never completed, are new-proposed, or that are the subject of “paper studies” (cost projections for the latter).

This data was used (see Preface to Reactor Modules) with a proposed methodology to incorporate the large variations in the designs and their state of development towards a final NOAK power plant that is optimized for commercial operations. This resulted in a small, but arguably more defensible basis for the values included in the What-It-Takes table for overnight capital cost.

Table R2-1. Historical capital costs for completed fast reactor systems and projected costs for recently announced or currently under construction systems.[1]

Reactor and Size	Total Capital Cost (2006 $)	Specific Capital Cost ($/kWe)
MONJU (Japan) 280 MWe (completed)	$6B (2006 $)	$21,400/kWe (2006 $)
Superphenix (France) 1,240 MWe (completed)	9B Euros = $11B	$8,870/kWe (2006 $)
Proposed Large Japanese Sodium-cooled Fast Reactor 1,500 MWe (announced proto)	$2.3B (all-in costs)	$1,600/kWe (all-in costs)
BN-800 (Russia) (under construction) (2007 estimate) Revised 2008 estimate from (Proatom 2008) reflecting schedule slippage (Platts 2009) and procurement difficulties. Completion date has slipped from 2010 to 2014.	$2B in 2006 $	$2,500/kWe $ >$6000/kWe (2008$)
Future French Prototype (Mainichi Daily News 2006) 800 MWe (announced proto)	1.5B Euros ($2.0B in 2007 $)	$2,500/kWe
Kalpakkam Prototype FBR (India) (Subramanian 2006) (under construction)	$767M	~$1,500/kWe

1. There is not sufficient and publicly-available “lessons learned” information to explain the above historical costs. Historical costs are usually “all-in” or total capital cost and include financing and owner’s costs. Announced and “under construction” projects are generally expressed as overnight costs in constant dollars. U.S. standard GDP deflators were used to escalate historical costs to 2006 $. Most of these costs appear in the references listed at the end of this section.

Table R2-2. Projected capital costs for never-completed, new-proposed, or conceptual fast reactor systems. (Data compiled in 2006, the new and proposed projects have certainly been affected by the increases in commodity and labor pricing from 2006 to 2008. Where available, new data is cited in the text to follow.)

Facility Name	Proposed Location	Size/Capacity (Electric and Thermal)	Units	Proposed Operation dates	Capital overnight cost in millions of local currency units	Currency type (FCU) Foreign Currency Unit	Exchange rate to $ ($/FCU)	Exchange rate date	Capital cost in "then year" $M	Deflator	Capital cost in 2006$M	Electricity Specific Overnight Capital Cost ($/kWe)	Thermal Energy Specific Overnight Capital Cost ($/kWth)
Clinch River Breeder Reactor Project	Oak Ridge, TN USA	1000 350	MWth Mwe	late 1980s (term in 1983)	3600	US$	1	1984	3600	1.75	6300	18000	6300
BN-800	Beloyarsk, Russia	2300 800	MWth Mwe	2010	2000	US$	1	2006	2000	1	2000	2500	870
Prototype Fast Breeder Reactor (PFBR)	Kalpakkam, India	1400 500	MWth Mwe	2010	3492	crores	0.205	2003	717	1.07	767	1534	548
Japanese Sodium-cooled Fast Reactor (JSFR)	Japan (Conceptual Plant)	3530 1500	MWth MWe	2050	224700	Yen	0.009346	2006	2100	1	2100	1400	595
General Electric S-Prism (modular:2 power blocks/4 Rx modules) Nth of a kind	USA (Conceptual Plant)	4360 1651	MWth MWe	2020	2200	US$	1	1996	2200	1.22	2684	1626	616
Low Conversion ratio variants of PRISM	USA (Conceptual Plant)	1680	Mwe	2020				2004		1.047		1600-1700	
ANL-AFCI-118 Report (Hoffman)		4430	MWth										

R2-7. DATA LIMITATIONS

All fast reactors constructed to date have been "first-of-a-kind" (FOAK) facilities and typically a size substantially less than full scale commercial power plant that have not enjoyed the economic benefits (lower costs) of construction learning and near-design replication (FOAK to NOAK cost improvement) that, to some degree, thermal water reactors enjoy or the benefits of economy of scale. As a result, the specific capital cost for completed facilities is quite high. The projected specific capital cost given for the reactor cost estimates appearing in planning or "paper studies," is usually optimistic in that it incorporates some developer optimism. All this makes projection to the cost of a full scale NOAK commercial power plant quite uncertain. The upper bound of what-it-takes based on past experience is very high and the lower bound of what-it-takes suggest a potential for a significant cost savings over LWRs assuming the design improvements from the lessons learned and design optimization can be achieved as have been suggested (e.g., Boardman 1999).

Newer FR paper studies are incorporating many new innovative features that should lend technical support to what seem to be optimistic claims. An IAEA conference on "Fast Reactor Design with Emphasis on Economics" was held in Vienna in October 2008 (Williams 2009). Ideas for cost improvements were suggested, but no specific cost data were given. It is likely that such data are considered proprietary. The industrial participants (GE-Hitachi 2008) (Energy Solutions 2008) (AREVA 2008) in the former U.S. Global Nuclear Energy Partnership (GNEP) activity produced reports describing their concepts for new fuel cycles in the U.S. Each of these reports suggested some sort of prototype fast reactor to demonstrate transmutation, but again no detailed cost information was available, and none of these companies detailed how the $/kWe cost would decrease in going from the prototype to the FOAK to the NOAK units.

R2-8. COST SUMMARIES

As can be seen above, specific capital costs, both realized and projected, for fast reactors vary widely. Cost experience for actual projects has not been good. These systems have additional piping and components than for LWRs because of the additional intermediate coolant loop (water to sodium) and the larger equipment needed to pump and handle liquid sodium. As part of the Generation IV program, however, new design concepts are being investigated that will hopefully include much enhanced passive safety, simpler systems, and improved economics. The Japanese have worked on such a concept, the JSFR, which they believe for an NOAK system can come in at well below $2,000/kWe including interest during construction (Ono et al. 2007). Recent PRISM studies (Ehrman and Boardman 1995; Dubberly et al. 2003a; Dubberly 2003b; Fletcher 2006; Forsberg 2007) for multiunit modular plants are also calculating specific capital costs in or below this $2000/kWe range. Based on the large projected increases in commodity and labor costs from 2003 onward (discussed in more detail in Module R1 for thermal reactors), this cost range is no longer considered valid.

As mentioned earlier, many nuclear critics believe that fast reactors will have inherently larger costs than LWRs. Russian experience has shown this factor to be more like 60% (VVER cost versus BN cost) (Minkov et al. 1990). At the 2008 IAEA meeting (Williams 2008), the Russian representative suggested that at a unit size of 1800 Mwe or larger, the $/kWe cost of a fast reactor system should be equivalent to or smaller than that of their LWR (VVER) reactor systems. It should also be noted that estimates prepared by designers of the EFR show it to be a 25% cost increase per kilowatt than the European Pressurized-water Reactor (EPR), also estimated by the same team. These cost comparison are currently speculative because neither the LWR nor the fast reactor have been built in the developed world in the past 2 decades to furnish much actual data for comparison. Again, as mentioned earlier, the fast reactor has benefits that the LWR does not, namely the ability to either breed or burn actinide materials, and in contributing to closing the fuel cycle, while generating electricity on the grid and eliminating the need for enrichment.

The module cost information is summarized in the What-It-Takes (WIT) cost summary in Table R2-3. These values are based on cost analyst's judgment and are intended to provide a cost distribution that is consistent with LWR (Module R1) values and the sparsely available cost data for commercial-scale NOAK fast reactors. The primary driver is the view that the lowest cost estimates of concepts are probably too optimistic and the extrapolation of smaller scale demonstration projects is too pessimistic, so the range of cost estimates is reduced. As is to be expected, with greater uncertainty a broader range compared to the LWR is appropriate which is represented by a slightly lower upside and a somewhat higher downside. Future versions of this report are anticipated to provide greater detail and better basis and justification for the cost values per the methodology described in Module RP2. The summary shows the reference cost basis (constant year U.S. $), the reference basis cost contingency (if known), the cost analyst's judgment of the potential upsides (low end of cost range) and downsides (high end of cost range) based on references and qualitative factors. These costs are subject to change and are updated as additional reference information is collected and evaluated, and as a result of sensitivity and uncertainty analysis. Refer to additional report for additional details on the cost estimation approach used to construct the WIT table (Table R2-3).

Table R2-3. What-It-Takes cost summary table from 2012 AFC-CBD Update.

What-It-Takes (WIT) Table				
Reference Cost(s) Based on Reference Capacity	Upsides (Low Cost)	Downsides (High Cost)	Expected (Mean Cost)	Selected Values (Nominal Cost)
Overnight Cost for NOAK FR in U.S.	$2200/kWe	$7000/kWe	$4600/kWe	$4600/kWe
O&M Fixed Component including D&D fund contribution (no ref. available)	$60/kWe-yr	$85/kWe-yr	$70/kWe-yr	$65/kWe-yr
O&M Variable component including Capital Replacement Component (no ref. available)	1.0 mills/kWh	2.7 mills/kWh	1.9 mills/kWh	2.0 mills/kWh

O&M costs for the reactor have been included in this edition of the Cost Basis Report. They are applied in the same manner as in Module R1—they have a fixed and variable component. The code of accounts structure would also be the same as that described in Module R1. O&M costs are expected to be somewhat larger for FRs as compared to LWRs, mainly because of the more complex systems. In the table above the fixed component of the O&M cost has been increased somewhat from the 2008 values to reflect O&M cost escalation. The following Table R2-4 updates all costs to Year 2015$.

Table R2-4 What-It-Takes cost summary table updated to 2017$.

What-It-Takes (WIT) Table				
Reference Cost(s) Based on Reference Capacity	Upsides (Low Cost)	Downsides (High Cost)	Expected (Mean Cost)	Selected Values (Mode Cost)
Overnight Cost for NOAK FR in U.S.	$2400/kWe	$7600/kWe	$4700/kWe	$4100/kWe
O&M Fixed Component including D&D fund contribution (no ref. available)	$65/kWe-yr	$92/kWe-yr	$78/kWe-yr	$76/kWe-yr
O&M Variable component including Capital Replacement Component (no ref. available)	1.1 mills/kWh	2.9 mills/kWh	2.1 mills/kWh	2.2 mills/kWh

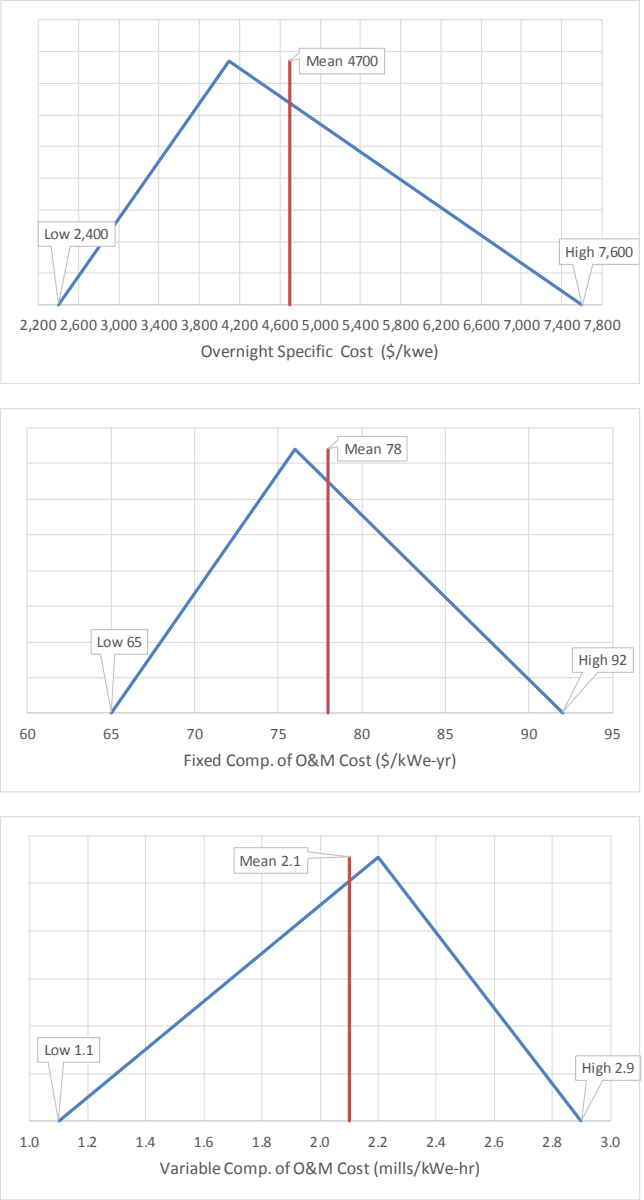

Figure R2-2 Probability Distributions for SFR Cost Parameters.

R2-9. SENSITIVITY AND UNCERTAINTY ANALYSES

No studies of this type have been undertaken recently. It is known, however, that as with thermal reactors, the factors that will most influence the levelized unit electricity cost are the reactor capacity factor (% of time it is generating electricity), the total capital cost, and the time it takes to construct it.

R2-10. BIBLIOGRAPHY

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Module R3

Gas-Cooled Reactors
(High-Temperature Reactors)

Module R3

Gas-Cooled Reactors (High-Temperature Reactors)

R3.MD SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2012 AFC-CBR)
- **Estimating Methodology for latest (2012 AFC-CBR) technical update from which this 2017 update was escalated:** There are a number of design/cost studies and limited operational experience that were accessed to try to project the overnight capital cost of gas cooled reactors.

R3.RH REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2009 as Module R3.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2012
- **New technical/cost data which has recently become available and will benefit next revision:**
  - A new methodology being considered for estimating costs of new advanced reactor concepts considers cost scaling based on different design features (e.g., whether the technology scales based on thermal or electric power). The method in 2015 RP-2 and now a Support Document for the 2017 Version (*A Proposed Methodology for Transformation of Reactor Cost Data to the "What-It-Takes" Table*, See SD6) can be applied to the data available for small-scale single unit demonstration projects.
  - Need to incorporate historical and projected information on O&M costs to complete section.
  - Consider renaming this Module to Gas-Cooled Thermal Reactors.

R3-1. BASIC INFORMATION

2009 AFC-CBR Introduction. This module was a new addition to the 2009 Advanced Fuel Cycle report cost database. Gas-cooled reactors have been operated, mainly in the UK, for many years. Designs of current interest take advantage of the higher coolant temperatures available with gas-phase (in this case helium) cooling and the associated higher thermodynamic efficiencies. For this reason the moniker "High-Temperature Reactors" (HTR) is more appropriate. It has been added because of increasing interest in this type of reactor for process heat applications, especially where process heat in the range of 700 to 900°C is needed. Nearly all proposed HTR designs for large-scale applications (as opposed to very small space-related applications) are thermal neutron spectrum reactors and use graphite as the moderator. Also, nearly all of today's HTR designs are modular in nature (i.e., a plant would consist of multiple reactors of 200 MWth to 600 MWth in capacity). Some of the generic economics of smaller reactors is discussed under "Modularity" in the Main Report; however, because of the considerable interest in this particular reactor type, a separate module designation and section were dedicated to HTRs in this report. In addition to versatility for process heat applications, proponents of HTRs stress the safety of this design, since the graphite-embedded particle fuel cannot melt, and the reactivity decreases with temperature. The tri-structural isotropic (TRISO) fuel itself is designed such that fission products cannot easily escape outside the fuel particle coatings; hence, the fuel itself is part of the "containment." The higher temperature variation of the reactor type, the very-high temperature reactor (VHTR), is one of the six concepts being

developed under the International Generation IV Reactor Systems Program. In the early 2000's, the U.S. was developing this reactor type as the Next Generation Nuclear Plant (NGNP), which was selected as first priority of the U.S. Generation IV Program. Currently the commercialization effort for this project is oriented toward high-temperature process heat rather than electricity.

2012 Update AFC-CBR Introduction. Module R3 of the 2009 AFC-CBR described in detail the technical and economic considerations associated with Gas-cooled Reactors. (The name of this Module has was changed to High Temperature Gas-cooled reactors to reflect the importance of the high temperature for industrial applications and the implied use of high-temperature TRISO-type fuel. (Note that solid-fueled salt-cooled reactors can also achieve high temperatures. These will be discussed in Module R8). The HTGR has garnered considerable renewed industry and government attention over the last twelve years because of its potential to produce process heat for a multitude of industrial uses, including hydrogen production. More recently HTGR-generated heat has been proposed for use in the petrochemical industry; however, the recent low prices for wholesale natural gas are making nuclear heat appear less attractive. Electricity generation (or co-generation of heat and electricity), however, is still an important mission. The Department of Energy and the Nuclear Industry at one time were spending millions of dollars per year toward HTGR development, mainly in the planning for a demonstration reactor, the Next Generation Nuclear Project (NGNP). Pre-conceptual designs for both pebble-bed fueled and prismatic fueled reactors have been prepared by nuclear industry vendors with DOE support. Current thinking was that the NGNP demo would be in the 350 to 600MW (thermal) range, and that process heat would be the main product. This demonstration plant would need to be located near an industry that could purchase the energy and help offset the life cycle costs. (Discussion of the technology and economics of TRISO-type particle fuel appears in the update for Module D1-3. There is also a discussion therein of the status of national HTGR and TRISO fuel programs outside the US.) Presently Japan and China are the only countries with operating demonstration projects. China's HTR Program plans for the ultimate deployment of 18 modular pebble-bed HTGRs of 110MWe each. It should be noted that a U.S. company, X-Energy, is pursuing R&D on this option.

This 2015 AFC-CBR does not contain any new background information; however, new selected values for low, mode, high, and mean (along with probability distributions) are presented. These values are presented in year 2015 dollars calculated by application of the appropriate escalation factor.

Economic Terminology. The economics of HTRs is generally expressed with the same types of figures of merit as for the thermal water reactor (R1) and fast (R2) systems, the main exception being that the unit cost of process heat (thermal kilowatts of millions of BTUs) is calculated rather than the unit electrical kilowatt cost. The HTR provides process heat that can be used for a wide variety of applications, only one of which is electricity. The energy supplied to the industrial applications that have been and are being evaluated is in the forms of electricity, steam, and high-temperature gas. For the purposes of analyses, these products are priced in terms of $/MMBtu (million BTU) required to generate them. This metric allows a direct comparison with the costs of the fossil fuel-based (e.g., coal, natural gas) products using conventional processes. (Wholesale natural gas prices have varied from 3 to 13 $/MMBTU over the last 10 years and are now in the low end of this range.) This is the target market for the HTR; the pure electricity supply market is a secondary market in which, for reference, the HTR is very competitive in niche applications because of the high net efficiency of the plants when compared with other nuclear technologies and similar net efficiency when compared with coal and natural gas fired plants.

The fuel cycle cost, including preparation of the TRISO fuel, for a HTR is just one of the four main components of the busbar levelized unit electricity cost (LUEC) from a nuclear power plant. ("Busbar" cost refers to the fact that the electricity cost is measured at the reactor plant boundary connection on the primary side of the switchyard transformer and does not include distribution [transmission] or other utility overhead costs.) The LUEC is usually expressed in mills/kWh or $/MWh; the value is the same in these

two units. (One mill=1/1,000th of a dollar or 0.1 cents). This and other economics-related definitions are described in the *Cost Estimating Guidelines for Generation IV Nuclear Energy Systems* (EMWG 2007). The four components of the LUEC are:

1. Capital component. Recovery of reactor capital plus financing costs. The capital component includes all "up-front" costs prior to commercial operation, including design, licensing, construction, project management, ownership costs, interest during construction, and reactor start-up (commissioning). This component of the LUEC also includes the returns to the investors made during plant operations, such as the interest portion of capital recovery.
2. Operations and maintenance component. Annual nonfuel costs including manpower, nonfuel consumables, and overheads. Manpower costs for refueling outages are usually captured in this category. Replacements for major capital items not related to life extension, such as steam generators, can also be placed in this category.
3. Fuel cycle component. The sum of the relevant costs for the needed fuel cycle steps (modules) converted to mills/kWh or $/MWh unit costs. Models such as G4-ECONS can perform this sometimes complex calculation (EMWG 2007), which involves both unit costs for fuel cycle steps and fuel cycle material balances. Depending on the utility, accounting practices, carrying charges (interest) on stored fuel, and fuel cycle materials undergoing processing are sometimes assessed to this category.
4. Decontamination and decommissioning (D&D) component. Usually covered by an escrow or sinking fund accumulated to cover D&D costs for the reactor at its end of life. The calculation of the levelized annual payments to this fund over the operational life of the reactor is described in EMWG's 2007 document.

R3-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

Most HTR designs intend to use a fuel consisting of TRISO coated particles embedded in a graphite matrix as discussed in Module D1-3 and helium as the primary coolant. For those producing electricity, a direct Brayton cycle is in the current generation designs; however, its demands on materials (suitable for high temperatures) and energy conversion equipment (direct helium drive turbine/generator) are more severe. Older HTR designs were operated on a steam (Rankine) cycle with a helium-to-water heat exchanger/steam generator. In an electric plant more than one reactor module could drive a turbine generator. Each combination of multiple modules and a T/G is called a "power block." Two or more "blocks" constitute a plant.

For the "process heat" plant the primary loop helium coolant will be pumped through a heat exchanger with the secondary side high-temperature coolant transported to the petrochemical, hydrogen, or other process facility. It should also be noted that some HTR concepts do not involve a gas coolant. The Advanced High Temperature Reactor (AHTR) concept involves the use of a molten salt as a coolant. The better heat transfer allows more power to be produced in a given size core as compared to gas coolant.

R3-3. PICTURES,SCHEMATICS AND DEPLOYMENT STATUS

Figure R3-1 shows a generic schematic for a Gas-Cooled Reactor System, in this case the VHTR being considered under the Generation IV Reactor program. For this diagram's example concept the heat is being used to drive thermochemical hydrogen production process rather than to drive a turbine/generator. More recently the VHTR mission has been redefined to supply process heat (or high-temperature steam) to more conventional petrochemical facilities and unconventional hydrocarbon recovery operations.

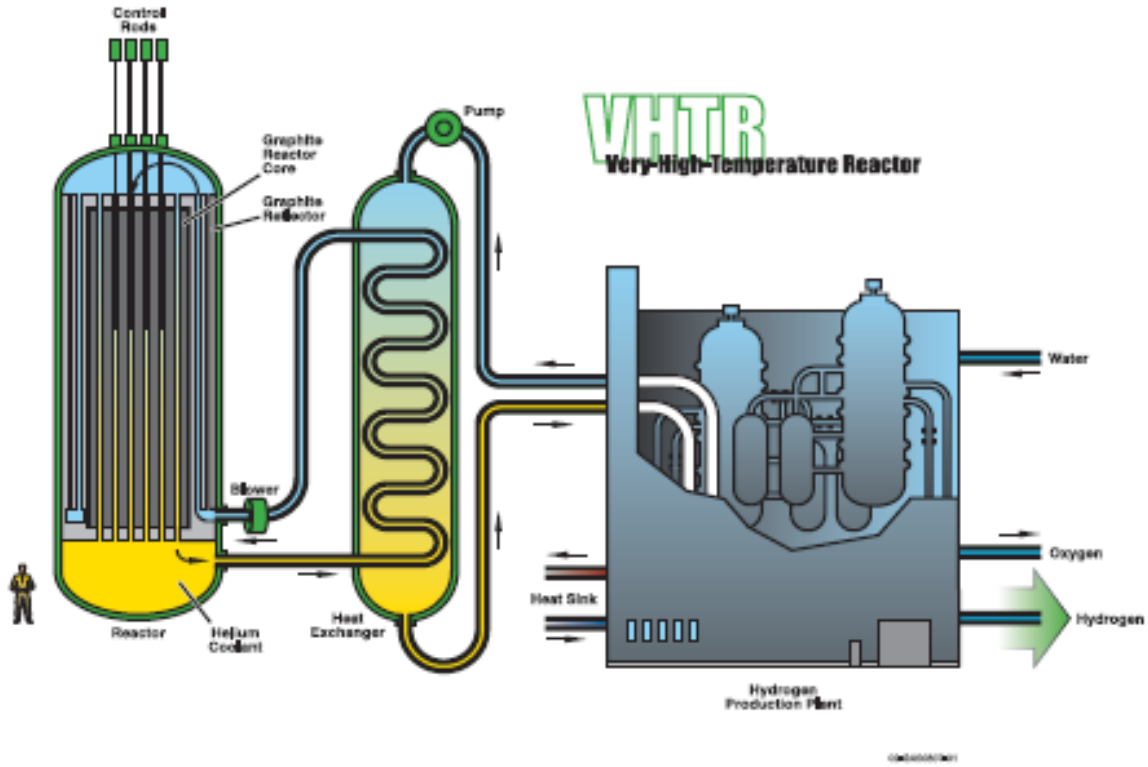

Figure R3-1. VHTR/HTR schematic.

The concept of HTRs is not a new one and has been manifested in many forms over the last 5 decades. The UK has a large, but aging fleet of gas-cooled reactors (MAGNOX and Advanced Gas-Cooled Reactors [AGR]) using CO_2 as the coolant and more conventional (non-particle) fuel. These reactors were built in the 1950s and 1960s, and the construction cost data are not particularly applicable to today's designs. One high-temperature gas-cooled reactor has been constructed and operated for electricity production in the U.S.: the Fort St. Vrain High Temperature Gas-cooled Reactor (HTGR). It was designed by General Atomics (GA) and utilized U and (U, Th) TRISO fuel produced in a small fabrication facility in Sorrento Valley, California. It was constructed in the 1970s and shut down due to operational difficulties in the early 1980s. Again, the construction cost data from that time would not be useful in gauging costs today. In the early 2000's, GA was still pursuing the HTGR design; however, the product was to be more modular in nature, called the Modular Helium Reactor (MHR). A direct cycle design was proposed because of the high thermodynamic efficiency possible (i.e., approaching 50%). This design has also been proposed as a "deep burn" actinide burning reactor, a weapons plutonium destruction reactor, and as a heat source for hydrogen production using high-temperature processes such as the High Temperature Steam Electrolysis (HTE) process. As part of the Generation IV program, the U.S. DOE was proposing a prototype HTGR called the VHTR (Figure R3-1 above). The actual demonstration project is called the Next-Generation Nuclear Plant (NGNP), which was described above. It will be oriented toward process heat applications requiring at least 750°C temperatures. U.S. reactor manufacturer consortia have proposed different VHTR/NGNP designs, and pre-conceptual designs have been prepared. None of the cost data associated with these designs has been made public; however, costs of around $4B are expected for a first-of-a-kind reactor. The quoted range of costs from the FY07 Preliminary Conceptual Design Report (PCDR) was $3.8B to $4.3B (2007$) for reactors between 500 and 565 MWth. The plants covered in that work were high temperature (900 to 950°C), which included intermediate heat exchangers with secondary helium loops supplying steam generators and hydrogen production facilities. As noted above, the focus has shifted to a steam plant supplying steam and

electricity in, for example, a co-generation application with an industrial facility or in recovery of unconventional hydrocarbons from oil sands or shale. The project costs for the revision in the focus have not been formalized but are anticipated to be in the upper part of the range cited in FY07.

As mentioned in Module D1-3, the TRISO fuel concept can be either prismatic (compact with embedded TRISO particles), or spherical (billiard-ball sized pebbles with embedded TRISO particles). China is currently pursuing the pebble-bed modular reactor (PBMR) concept. The PBMR concept was also evaluated in South Africa by the utility Eskom, but due to high costs and currency fluctuations the project was put on hold. Eskom cost estimates are highly proprietary; however, early speculation was that LUEC costs of less than 20 mills/kWh (including capital amortization) were considered possible. The current PBMR designs reflect a shift in focus from the high-temperature, higher power annular core and the direct Brayton cycle plant for electricity production to a lower temperature, lower power cylindrical core for production of steam and electricity using a Rankine cycle.

R3-4. MODULE INTERFACES

Front-end. The fuel for most concepts is the UO_2 or uranium-oxycarbide (UCO) TRISO particle fuel at enrichments of 8 to 19.9% U-235, thus keeping its fissile enrichment in the "LEU" range. Fabrication of this fuel is discussed in Module D1-3. Presently there is no large scale facility for fabrication of this fuel. What would be shipped to the reactors would be critically-safe packages of spherical pebbles or packaged graphite fuel blocks with embedded "prismatic" compacts.

Back-end. The discharged spent particle fuel will still be within its graphite matrices. The pebble-type spent fuel could be packaged in special barrels that because of decay heat would probably require some active cooling, possibly air or gas rather than water. The hexagonal blocks from the prismatic variety can also be packaged and stored. Another option for more compact storage would be to push the compacts out of the hexagonal graphite block and store them in a manner similar to discharged pebbles.

Reprocessing of TRISO spent fuel would be more difficult and less well developed than for light-water reactor (LWR) or fast reactor (FR) fuel because of all the fission product release barriers that were built intentionally into the fuel. To dissolve the fissile material, one must first destroy the graphite and the multiple coatings that constitute the TRISO particles. A few reprocessing schemes, including burning away the graphite and crushing the remaining UO_2 or UCO particles, have been suggested for this type of fuel.

Thorium. It should be noted that thorium-containing TRISO-type particles can be introduced into GCR systems to extend the burn-up. The fertile Th-232 is converted to U-233, which is itself fissile and can extend the life of the overall core. Use of thorium in GCRs has been demonstrated in test reactors in Europe and in the U.S. commercial unit at Fort St. Vrain, Colorado. Fuel fabrication issues associated with thorium are discussed in Module D1-8.

R3-5. SCALING CONSIDERATIONS

In general, the cost of electricity and the specific capital cost decrease with larger reactor size (electrical generation capacity). There is likely to be a point where factory production of small reactor modules, as opposed to traditional onsite construction, will allow reduction of unit costs. Studies, such as those being pursued by the Generation IV Economic Modeling Working Group, are investigating this issue.

R3-6. COST BASES, ASSUMPTIONS, AND DATA SOURCES

2009 AFC-CBR Discussion. The question often arises as to how the specific overnight cost of a HTR compares to that for LWRs or fast reactors. From a pure engineering standpoint, larger structures are required because of the lower power density of the HTR core as compared to water reactor cores and the requirement for high-temperature service. However, the higher thermodynamic efficiency of HTRs vis-à-vis water reactors should help to drive down the $/kWe cost, since more units of electrical capacity are available per unit of heat. Several nations are pursuing prototype HTRs such as China, Japan, and South Africa, but projected cost information is sketchy, and the prototypes for which cost estimates exist are necessarily the size of the eventual commercial unit that might be deployed. Because of recent price volatility in natural gas and the need to reduce "carbon footprints" and concerns with the security of feedstock for foreign sources, many industries are now considering HTRs as a reliable source of non CO_2-emitting process energy and heat with a stable price, (e.g., $/million BTUs) (Nuclear Engineering International 2009). These industries include petro-chemical industries, fertilizer manufacturers, refineries, oil sands and oil shale extraction, and upgrade companies in addition to chemical companies. Even though the early applications of the gas reactor technology focused on production of electricity, because of the interest by the process industries, the current focus is on supplying carbon-free process heat and energy.

Among the applications of the HTGR technology that have been studied worldwide are tar sands bitumen separation and upgrading, hydrogen production, synfuels production from coal, crude oil beneficiation, ammonia products, ethylene cracking, and steelmaking. These applications use energy in one or more of the forms of electricity, steam, high-temperature gas, hydrogen, and oxygen. The HTR technology can provide energy in all of these forms with stable cost and without emissions of greenhouse gases. The use of nuclear energy to reduce "carbon footprints" in these industries is one of the critical considerations when judging competitiveness of the HTR for these applications. The economics of these applications consider the need for security in the source of energy, the cost and stability of the cost of alternatives, as well as the potential price of carbon emissions. EPRI (2009) estimates for example that a $50/metric ton tax on CO_2 would increase the cost of electricity produced from a conventional coal plant by $43/MWh and from a conventional natural gas fired combustion turbine plant by $19/MWh. Allocation of costs and revenues between process heat and electricity is a complex issue and is discussed in two of the references for this section (Florido 2000 and EMWG 2007).

Consistent with other cost modules in the AFC Cost Basis report, cost data was collected on all types of HTRs, regardless of their timeframes for development. The data on these reactors were collected through various reference sources including the trade press and trade press sources. The cost data represent the costs for HTRs actually completed and proposed (projections made for paper studies). The cost data collection in Table R3-1 includes commercial units (e.g., Fort St. Vrain, Peach Bottom), as well as reactors developed for research purposes. Many of these reactors are first-of-a-kind or demonstration units and are not directly comparable. Further discussions on the limitations of this cost data are included in Section R3-7. All-in costs include financing and owner's costs in addition to the usual "overnight" costs unless otherwise noted.

Table R3-1. Historical capital costs for completed gas-cooled reactor systems and projected costs for recently announced, currently under construction, or hypothetical systems (2009 AFC-CBD).

Reactor and Size	Total Capital Cost	Specific Capital Cost ($/kWe)
Fort St Vrain (Colorado, USA) One 350 MWe unit (completed in late 1960s) (first-of-a-kind [FOAK])	$200M (1968$) all in $1.4B (2008$) (Costs escalated by historical Handy-Whitman Utility Construction Indices)	$606/kWe (1968$) $4303/kWe (2008$)
Japanese 30MW(th) HTTR at Oarai Research Center (test reactor: no electricity production) (IAEA 2007) (FOAK) (U.S. Handy-Whitman Index used for escalation)	$700M (1992$) $1260M (2008$)	$23,000/kWt (1992$) $41,000/kWt (2008$)
AVR (Arbeitsgemeinschaft Versuchsreaktor) Pebble-bed; Julich, Germany; 40 MW(th); 15 MW(e) 1966 (FOAK) (Van Heek 2009)	70M Deutschmarks (1966) ~to 17.5M (1966$) $144M in 2008$ (based on 1966 cost estimate and H-W escalation)	$1166/kWe in 1966$ $9600/kWe in 2008$
THTR (Thorium High Temperature Reactor); Germany 300MW(e) (FOAK) (Saunders 2006)	Original estimate $411M $825M in 2008$ Actual cost $2530M (1988$) $5000M in 2008$	$1370/kWe (1988$) $2750/kWe (2008$) $8430/kWe (1988$) $17,000/kWe (2008$)
10 MW(e) Chinese HTR-10 Pebble Bed; Tsinghua University (test reactor) (FOAK) (World Nuclear Association)	Not given in IAEA database	–
U.S. Peach Bottom-1; 40 MW(e); 1967–1974 (utility owned, GA designed) (FOAK)	Not given in Komanoff (1981) database of U.S. power reactor actual costs (Komanoff 1981)	–
Japanese conceptual design for 4 module plant of total capacity 1148 MWe (Shintaro 2001)	315B yen (2001 yen) $3.1B (2001$)	$2750/kWe (2001$) $3500/kWe (2008$)
GA (USA) pre-conceptual design for 4 module (1152 MWe total) MHR (GNEP 2008) (NOAK)	$1.57B (overnight cost in 2006$)	$1,639/kWe (overnight) (consistent with other GA studies on hydrogen, etc.)
10 module MIT PBMR design project (1100 MWe total), K. Williams reviewing author (Williams-G4ECONS 2009) (NOAK)	$2.3B (1992$) (all-in) $4.2B (2008$)	$1860/kWe (all in) $3990/kWe (all in)
MIT Study on Integration with Oil Sands projects (Bersak 2007)	–	$4000/kWe overnight for one 172 MWe module (2008$) $3333/kWe overnight for four modules (USA)
Indonesian PBMR study (Nasrullah 2008)	–	$2515/kWe (all-in)
4 and 8 module GA-design MHTGR production reactors with co-production of electricity (135 MWe per module raised to 175 MWe per module to account for technology improvement (NPR1991) (FOAK)	$3.56B incl dev't & contingency for 4 modules (1990$) $6.23B (2008$) $4.85B incl dev't & contingency for 8 modules (1990$) $8.49B (2008$)	$8900/kWe (all-in) $6060/kWe (all-in)
South African PBMR module (80MWe) (Creamer Media 2009) (FOAK)	$7B Rand including development costs to date. ($875M)	$10,900/kWe (all-in plus some development)
Proposed Kazakhstan 50MWe Gen IV HTGR Project (Nuclear Engineering International 2009)	$500M for Prototype	$10,000/kWe

2012 Update Discussion. As with LWRs and SFRs the most commonly found traditional cost figure of merit is the specific capital cost in dollars per kilowatt of electrical power capacity. Since most proposed applications of the MHTGR now are for process heat, however, the overnight cost figure of merit has now become the specific cost of thermal capacity in $/kW(th). Again one must be careful to understand whether the specific "overnight" cost or the specific "all-in" cost is being discussed. Since these modules are for comparing technology-related costs as opposed to financing-related costs, the "overnight" figure-of-merit is the applicable one discussed and presented here. If only the $/kW(th) figure-of-merit is given, one can approximate the $/kWe cost for an HTGR system by dividing the former thermal figure-of-merit by an assumed thermodynamic efficiency. The cost of a turbine/generator (T/G) must be added to the thermal only system prior to this adjustment (which would probably add less than 15% to the heat-only capital cost. Many HTGR designs will include the T/G so that they can be co-producers, therefore T/G costs are included.

The values shown in the following tables are mostly for Nth-of-a-kind (NOAK) projects unless otherwise noted and include the effects of assumed construction learning from the lead or "first-of-a-kind" (FOAK) or "lead" plants. Table R3-2 below presents HTGR specific overnight cost data gleaned from various literature sources since 2009. The table includes overnight capital cost data from two surveys (WNA, n.d.; Yankov, 2012) of country-by-country reactor specific costs and the results of two cost/design studies (INL 2011; INL 2012) supported by Idaho National Laboratory and DOE-NE. These latter studies are based on design work performed by nuclear industry subcontractors for the NGNP Program. Operations and maintenance costs for HTGR systems will be added to future AFC-CBR updates.

Table R3-2. Specific Overnight Costs for HTGRs from Literature Sources in the 2012 AFC-CBD Update.

Study or Ref /Year	Low Value $/KWe or $/KW(th)	Reference Value $/KWe or $/KW(th)	High Value $/KWe or $/KW(th)
AFC-CBR 2009 (NOAK - WIT)	3000	4500	7500
World Nuclear Association website on China (WNA, n.d.) 2 module HTR-PM at Shidaowan, China (Each module is 110 kwe; electricity prod.)	N/A	$3710 /KWe FOAK $1300/KWe target NOAK	N/A
Bulatom Presentation on SMRs: Chinese 2-module HTR-PM (Yankov 2012)	N/A	$3900/KWe (FOAK)	N/A
INL Technical Evaluation Study (INL 2011) (2010$) Lead & NOAKs NGNP 600 MW(th) NGNP 350MW(th) NOAK 4-pack [4 X600MW(th)] NOAK 4-pack [4 X 350MW(th)] (*43% thermodynamic eff., 850C reactor outlet temp)	 $4510/kW(th) $6550/kW(th) $1453/kW(th) ($3380/kWe*) $3113/kW(th) ($7240/kWe*)	 N/A N/A N/A N/A	 $9664/kW(th) $14036/kW(th) $1785/kW(th) ($4151/kWe*) $3824/kW(th) ($8890/kWe*)
INL Industrial Applications Study (INL 2011) 1-2 600MW(th) modules 3 or more 600MW(th) modules NOAK (*40% thermodynamic eff., 700-825C reactor outlet temp.)	 N/A N/A	 $2000/kW(th) ($5000/kWe**) $1400/kW(th) ($3500/kWe**)	 N/A N/A

The INL Technical Evaluation Study was the most recent, most valuable, and most detailed study accessed. The estimate was largely "bottom-up" in its preparation, and a contingency range of -30% to +50% was suggested to be used with the baseline estimates. This study presented only the $/kW(th) figure-of-merit. The author of this module assumed a thermodynamic efficiency appropriate for the reactor coolant outlet temperature of 850C in order to convert them to $/KWe.

R3-7. DATA LIMITATIONS

All helium gas-cooled reactors constructed to date have been first-of-a-kind (FOAK) facilities that have not enjoyed the economic benefits (lower costs) of construction learning and near-design replication (FOAK to NOAK cost improvement) that, to some degree, thermal water reactors enjoy. Instead, the specific capital ($/kWe) costs for completed facilities, which have been prototype units, have been quite high. The projected specific capital cost given for the reactor cost estimates appearing in planning or "paper studies" is usually optimistic in that it incorporates some developer optimism, but it may not include financing costs. Prototype and development costs are often left out of electricity-related costs, such as the typical "recoverable" $/kWe or LUEC, since the government rather than the utility may pay for these.

Newer HTR paper studies discuss the incorporation of many new innovative features that should lend technical support to the development of lower cost estimates. An International Atomic Energy Agency (IAEA) Technical Meeting on "HTGR Economic Analysis" was held in Washington, DC in October 2008 (Williams 2008). Ideas for cost improvements were suggested, but very little specific cost data were given. It is likely that a great deal of such data is considered proprietary. Most of the meeting dealt with economic modeling issues and the use of G4-ECONS and other models to calculate the LUEC. One industrial consortium participant (GA, CH2M-Hill, KAERI, Hamilton Sunstrand, LISTO, and Potomac Communications) in the former U.S. Global Nuclear Energy Partnership (GNEP) activity recently produced a report (GNEP 2008) describing its concepts for new fuel cycles in the U.S. using LWRs, FRs, and HTRs in a symbiotic fashion. The emphasis was on actinide burning (transmutation), but again no detailed cost information was available, and the consortium did not describe how the $/kWe cost would decrease in going from the prototype to the FOAK to the NOAK ($1639/kWe) HTR "burner" units.

In September 2009 the DOE issued DE-FOA-0000149 for completion of design activities for high temperature gas reactor (HTGR) plants under Phase 1 of the NGNP Project.[a] This work should be completed by the end of FY10. An outcome of this work will be updates and improved confidence in the estimates of cost for deployment of the HTGR technology.

R3-8. COST SUMMARIES

2009 AFC-CBD Summary. The module cost information is summarized in the What-It-Takes (WIT) cost summary in Table R3-2. These values are largely based on cost analyst's judgment and are intended to provide a cost distribution that is consistent with LWR (Module R1) and FR (Module R2) values and the very sparse available cost data for commercial-scale NOAK fast and gas-cooled reactors. Future versions of this report are anticipated to provide greater detail and better basis and justification for the cost values. The summary shows the reference cost basis (constant year U.S. $), the reference basis cost contingency (if known), the cost analyst's judgment of the potential upsides (low end of cost range) and downsides (high end of cost range) based on references and qualitative factors, and selected nominal costs (judgment of the expected costs based on the references, contingency factors, upsides, and downsides).

a. FINANCIAL ASSISTANCE FUNDING OPPORTUNITY ANNOUNCEMENT, U. S. Department of Energy Idaho Operations Office, Next Generation Nuclear Plant Program – Gas Cooled Reactor Design and Demonstration Projects, Funding Opportunity Number: DE-FOA-0000149, Announcement Type: Initial, CFDA Number: 81.121, Issue Date: 09/18/2009.

These costs are subject to change and are updated as additional reference information is collected and evaluated, and as a result of sensitivity and uncertainty analysis. Refer to the main section of this report for additional details on the cost estimation approach used to construct the WIT table (Table R3-2). It should be noted that the selected nominal value of $4500/kWe (overnight) for the HTR compares to the same value of $3500/kWe (overnight) for the LWR (Module R1). It should be noted that escalation in commodity, labor, and procurement costs from 2003 through 2008 have caused the estimates $/kWe to be significantly increased for two classes of reactors (thermal in Module R-1, fast in Module R-2, and would presumably affect other reactor types in new Modules R-3 and R-4). A cost of $2000/kWt has been used in recent evaluations[b] for the overnight costs for an HTGR—at a nominal net efficiency of 42% (Rankine cycle) this is slightly more than $4,500/kWe. Interest during construction (IDC) and owners costs can add ~34% to this to a little over $6,300/kWe all-in cost. Table R3-3 lists these costs in this range.

Table R3-3. 2009 AFC-CBD What-It-Takes cost summary table.

What-It-Takes (WIT) Table			
Reference Cost(s)	Upsides (Low Cost)	Downsides (High Cost)	Selected Values (Nominal Cost)
Overnight Cost for NOAK HTR in U.S. $4500/kWe based on composite of various studies	$3000/kWe	$7500/kWe	$4500/kWe (NOAK)
Total Capital (all-in)	$3400/kWe	$9000/kWe	$4900/kWe (NOAK)

Operation and Maintenance costs for the reactor have not been included in this edition of the Cost Basis Report, since very little data is available.

Update AFC-CBD Summary. The unit cost values for the What-it-takes Table for the 2012 AFC-CBR have changed somewhat from those in the 2009 AFC-CBR. The values selected are based partly on Table R3-2 above with conversion of the specific cost figure-of-merit from thermal energy to electrical energy. The upside (low) value assumes that Far Eastern factories and workers might be able to produce and install HTGR-SMRs at a target NOAK cost of $2500/kWe which is above the stated Chinese target value of $1300/kwe, which was felt to be overly optimistic and probably reflected very early estimates and low labor and commodity costs. The high value was chosen to approximate the higher values derived from the INL Technical Evaluation Study (INL 2012). The nominal value is near the middle of the "high/low" range Note that this "What-it-Takes" overnight specific cost range is higher than for LWRs and SFRs. The realized higher projected cost for HTGR electrical generation systems is tending to push the HTGR mission toward thermal energy production only rather than electricity production. The high reactor coolant outlet temperature uniquely available from high-temperature reactors allows process heat applications that could not be undertaken by LWRs and SFRs with their lower reactor coolant outlet temperatures. If the price of conventional fossil process heat fuels such as natural gas rises again or "carbon-costing" is mandated, nuclear heat may become competitive in terms of the levelized unit cost of thermal energy ($/kw(th)-h or $/million BTUs).

Table R3-4. 2012 AFC-CBD Update: What-It-Takes Capital and Recurring Costs for NOAK HTGRs (2012$).[1]

	Upsides (Low Cost)	Selected Value (Nominal Cost)	Downsides (High Cost)
NOAK Overnight Cost ($/KWe)	2500	5000	8000
1. For uncertainty analyses a triangular distribution should be used with the values in this table.			

b. Based on personal communications with the NGNP Project.

For this 2017 edition of the cost basis report, the most recent specific cost data from 2012 has been escalated to 2017$ using a factor of 1.088 and some rounding, as shown below in Table R3-5. A triangular relative probability distribution with low, mode, and calculated mean value are displayed in Figure R3.2

Table R3-5. Year 2017$ What-It-Takes Capital and Recurring Costs for NOAK HTGRs (2017$).[1]

	Upsides (Low Cost)	Downsides (High Cost)	Excepted (Mean Cost)	Selected Value (Mode Cost)
NOAK Overnight Cost ($/KWe)	2700	8700	5600	5400
1. For uncertainty analyses a triangular distribution should be used with the values in this table.				

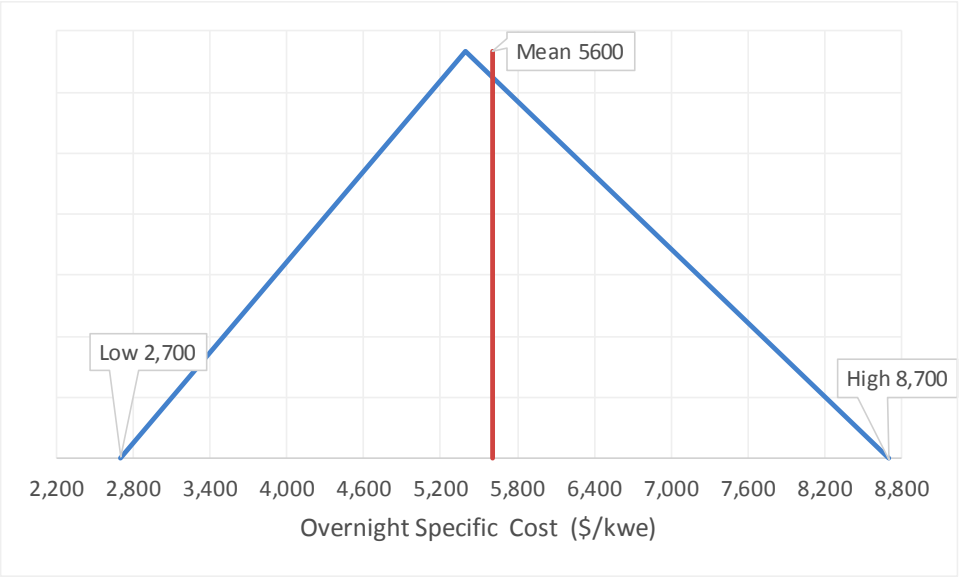

Figure R3.2 Probability Distribution for Specific Overnight Cost of HTR

R3-9. SENSITIVITY AND UNCERTAINTY ANALYSES

No studies of this type have been recently undertaken. However, it is known that as with thermal reactors the factors that will most influence the LUEC are the reactor capacity factor (% of time it is generating electricity), the total capital cost, and the time it takes to construct it. A recent Entergy study cited on a Power Technology Web site (Power Technology 2009) indicates that heat from HTRs can be competitive with heat from natural gas if the natural gas price climbs above $8/MMBTU[c]. No data on electricity costs were found from this summary of the Entergy study.

c. Note that this price includes no carbon tax on emissions from combustion of natural gas.

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Module R4

This module has been deleted and is no longer in use.

Module R5

Pressurized Heavy Water Reactors

Module R5

Pressurized Heavy Water Reactors

R5.MD SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2012 AFC-CBR)
- **Estimating Methodology for latest (2012 AFC-CBR) technical update from which this 2017 update was escalated:** There is a large amount of cost data for PHWRs built outside the U.S. that was compiled to estimate the overnight capital costs. The O&M costs were assumed to be the same as LWRs.

R5.RH REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2012 as Module R5.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2012
- **New technical/cost data which has recently become available and will benefit next revision:**
  - There should be sufficient O&M data available to improve the basis for those parameters. However, given the lack of interest of this type of reactor in the U.S., it doesn't seem warranted for the next revision.

R5-1. BASIC INFORMATION

This module describes the pressurized heavy water reactor (PHWR). These are thermal spectrum reactors that use "heavy" water—water enriched in deuterium—instead of "light" water as the moderator and primary coolant. All operating commercial reactors in the United States are light water reactors, but worldwide PHWRs currently provide approximately 25.6 GWe, with another 5.1 GWe under construction. PHWRs are in commercial use in Argentina, Canada, China, India, Pakistan, Romania, and South Korea. The existing reactors range in size from the 90 MWe Rajasthan Unit 1 in India to the 878 MWe Darlington units in Canada, while the units under construction range from the 620 MWe Cernavoda units in Romania to the 692 Atucha Unit 2 in Argentina (Nuclear News 2012).

The most common PHWR in commercial use is the CANDU, short for CANada Deuterium Uranium reactor (and a registered trade name owned by Atomic Energy of Canada, Limited). Nearly all PHWRs are CANDU-based reactors, so PHWR and CANDU are generally interchangeable terms when discussing PHWRs.

R5-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

Pressurized heavy water reactors share many characteristics with pressurized (light) water reactors. However, there are a few important differences. While these differences do not necessarily affect the cost of construction, they do affect the cost of operations.

The first difference is the makeup of the primary system. In PHWRs, the primary system (and thus the moderator and primary coolant) is heavy water; this is water enriched in the deuterium isotope of hydrogen. By enriching the water in deuterium, this slightly changes some thermal hydraulic properties; for example, the density increases from 1 g/cc to 1.11 g/cc (and is thus "heavier"). However, the main

effect is in its properties as a neutron moderator: heavy water has a smaller absorption cross section than light water, and thus removes fewer neutrons from the core. This leads to the second difference.

Since the moderator absorbs fewer neutrons, the PHWR can use lower-enriched uranium than the LWR. In fact, the PHWR can use natural uranium as its fuel, but the low fissile content (0.7%) of natural uranium leads to low fuel burnup before discharge—the typical PHWR burnup is 7 GWtd/MTU. The PHWR can be fueled using slightly (approximately 1%) enriched uranium to increase the discharge burnup to over 20 GWtd/MTU (Hindu 2012), but this is still less than the nearly 50 GWtd/MTU for PWRs. This low burnup leads to low resource utilization, which then causes high fuel throughput and spent fuel storage requirements. Another difference is a consequence of this requirement.

Since the reactor requires high fuel throughput, PHWRs are designed to accommodate on-line refueling instead of batch refueling. This allows the reactor to operate at full power while fresh fuel is inserted and used fuel is removed. LWRs generally have a pressure vessel, but the PHWR has pressure tubes within the core (called a calandria). The fresh fuel is inserted into the front of the pressure tube, pushing the used fuel out the back of the pressure tube. The PHWR can consistently achieve capacity factors around 90% (CNS 2012).

A PHWR typically uses unenriched (i.e., natural) uranium oxide pellets as its fuel. However, a PHWR can use other heavy metals as fuel. This allows the PHWR to be a part of any type of fuel cycle, ranging from single-use fuel through full recycle of actinides from used nuclear fuel and including uranium-233 breeding in a thorium cycle. The PHWR can also use the slightly-enriched recovered uranium (SEU) from the LWR fuel cycle.

While the PHWR fuel is similar to LWR fuel in chemical composition, it does not resemble the LWR fuel in form. The PHWR fuel is manufactured in short bundles instead of long assemblies, and these bundles are used in the online refueling process.

R5-3. PICTURES, DIAGRAMS, AND DEPLOYMENT STATUS

Diagrams of a typical PHWR, its fuel configuration, the associated fuel cycle, and fuel bundles are shown below in Figures R5-1 through R5-4.

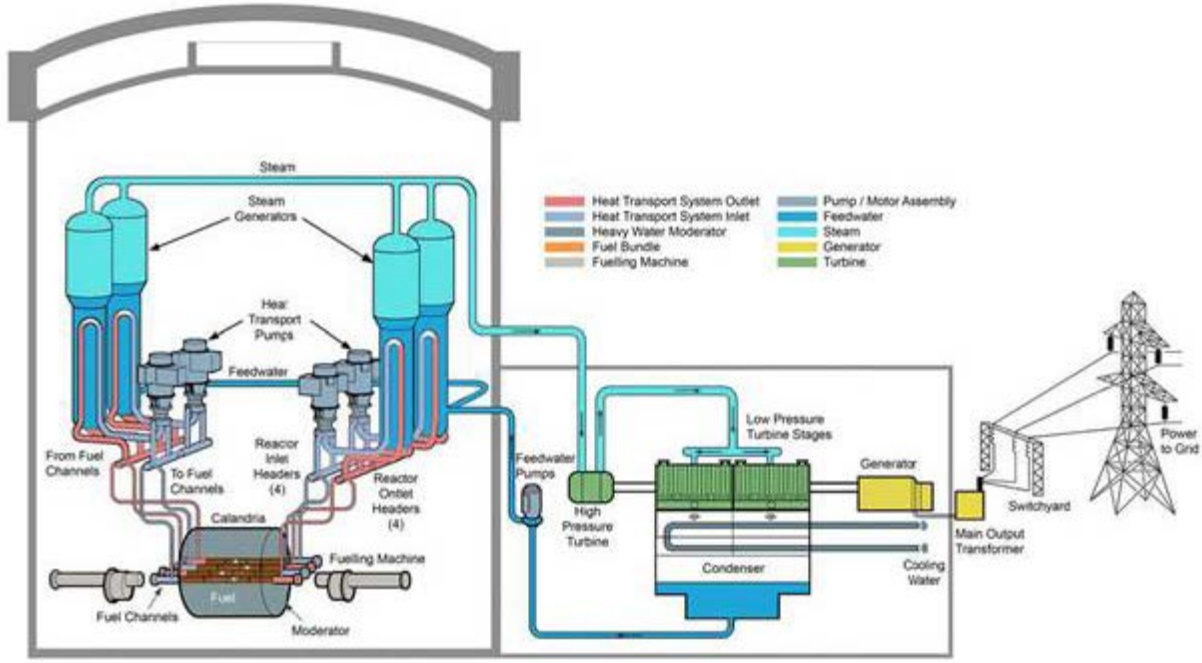

Figure R5-1. PHWR Schematic (IAEA a).

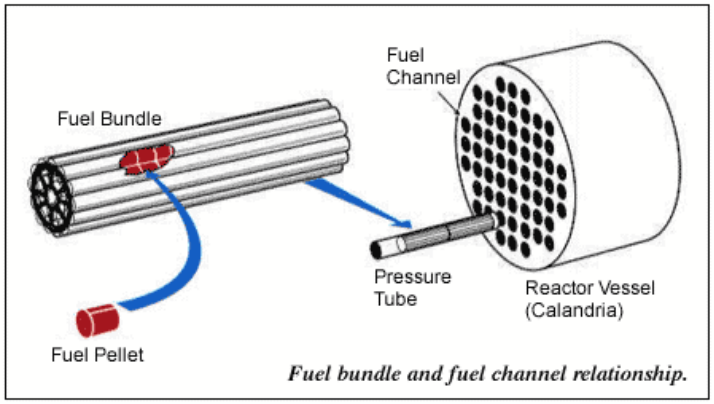

Figure R5-2. PHWR Fuel Pellets, Bundles, and Channels (Nuclear Tourist 2012).

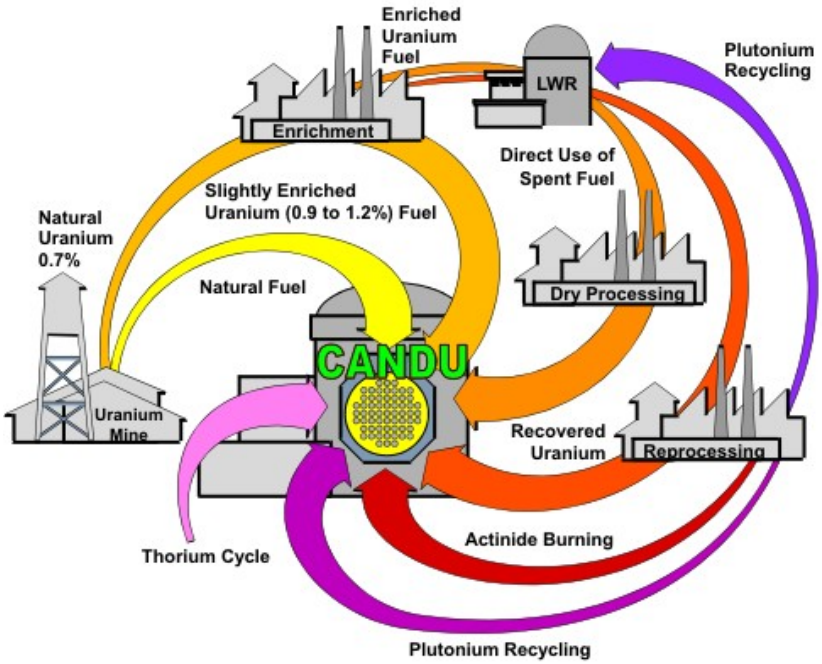

Figure R5-3. PHWR Fuel Cycles (Wikipedia 2012).

Figure R5-4. PHWR Fuel Bundles (WNA a).

The PHWR is deployed in seven countries, with the oldest operating units coming online in 1971 at Canada's Pickering plant. There are eight units currently under construction. Table R5-1 lists the currently operating and planned PHWRs. "UC" denotes "Under Construction".

Table R5-1. Currently Operating and Planned PHWRs (Nuclear News 2012).

Plant	Unit	MWe	Country	Date
Atucha	1	335	Argentina	1974
Atucha	2	692	Argentina	UC
Embalse	1	600	Argentina	1984
Bruce	1	769	Canada	1977
Bruce	2	769	Canada	1977
Bruce	3	730	Canada	1978
Bruce	4	730	Canada	1979
Bruce	5	817	Canada	1985
Bruce	6	817	Canada	1984
Bruce	7	817	Canada	1986
Bruce	8	782	Canada	1987
Darlington	1	878	Canada	1992
Darlington	2	878	Canada	1990
Darlington	3	878	Canada	1993
Darlington	4	878	Canada	1993
Gentilly	2	635	Canada	1983
Pickering	1	515	Canada	1971
Pickering	2	515	Canada	1971
Pickering	3	515	Canada	1972
Pickering	4	515	Canada	1973
Pickering	5	516	Canada	1983
Pickering	6	516	Canada	1984
Pickering	7	516	Canada	1985
Pickering	8	516	Canada	1986
Point Lepreau	1	635	Canada	1983
Qinshan	III-1	650	China	2002
Qinshan	III-2	650	China	2003
Kaiga	1	202	India	2000
Kaiga	2	202	India	2000
Kaiga	3	202	India	2007
Kaiga	4	202	India	2011
Kakrapar	1	202	India	1993
Kakrapar	2	202	India	1995
Kakrapar	3	640	India	UC
Kakrapar	4	640	India	UC
Kalpakkam	1	205	India	1984
Kalpakkam	2	205	India	1986
Narora	1	202	India	1991
Narora	2	202	India	1992
Rajasthan	1	90	India	1973
Rajasthan	2	187	India	1981
Rajasthan	3	202	India	2000
Rajasthan	4	202	India	2000
Rajasthan	5	202	India	2010
Rajasthan	6	202	India	2010
Rajasthan	7	640	India	UC
Rajasthan	8	640	India	UC
Tarapur	3	490	India	2006
Tarapur	4	490	India	2005
Kanupp	1	125	Pakistan	1972
Cernavoda	1	650	Romania	1996
Cernavoda	2	650	Romania	2007
Cernavoda	3	620	Romania	UC
Cernavoda	4	620	Romania	UC
Cernavoda	5	620	Romania	UC
Wolsong	1	597	South Korea	1982
Wolsong	2	710	South Korea	1997
Wolsong	3	707	South Korea	1998
Wolsong	4	708	South Korea	1999

R5-4. MODULE INTERFACES

This module interfaces with Module D1-7 (CANDU Fuel) for the incoming unirradiated fuel material. It would then interface with the same modules as the LWR module for the storage, reprocessing, or disposition of its used fuel.

R5-5. SCALING FACTOR CONSIDERATIONS

The PHWR is similar to the LWR with respect to the costs associated with reactor size. The two reactor types share the same basic characteristics with respect to lower specific cost ($/kWe) for larger reactors and lower total capital at risk for smaller reactors.

Figure R5-5 shows the rated power for each reactor by its date brought online. The cluster at 2020 is the group of reactors currently under construction.

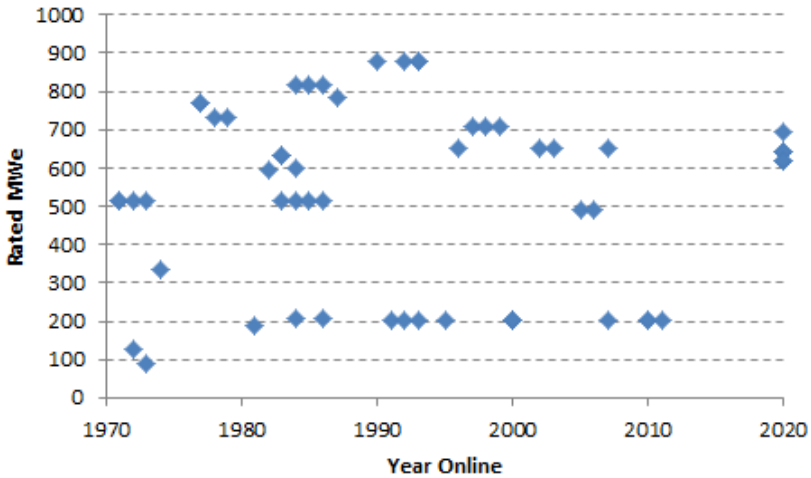

Figure R5-5. PHWR Rated Power versus Date Online.

The power range of currently-operating PHWRs is 90 to 878 MWe, a range spanning nearly 10x. However, the range for reactors coming online since 2000 is only 202 to 707 MWe (a range of 3.5x), and since 2010 only the 202 MWe reactors in India have come online. Conversely, all plants currently under construction (in Argentina, India, and Romania) are between 620 and 692 MWe.

R5-6. COST BASES, ASSUMPTIONS, AND DATA SOURCES

The most recent capital costs for PHWRs come from India, China, and Romania. The capital costs for those reactors are shown in Table R5-2.

Table R5-2. Recent Capital Costs for PHWRs.

Plant	Year	MWe	Indigenous Cost	Million USD	$/kWe	$/kWe (2011$)	Reference
Kaiga 1 and 2	2000	404	2896 (Crore)	630	1559	3414	Ramana a 2005
Rajasthan 3 and 4	2000	404	2511 (Crore)	546	1352	2961	Ramana a 2005
Qinshan III-1 and 2	2003	1300	--	2880	2215	4255	CNNC 2012
Tarapur 3 and 4	2006	980	6420 (Crore)	1396	1424	1723	Himal 2012
Cernavoda 2	2007	650	--	700	1077	1012	Chicago 2004
Rajasthan 5 and 6	2010	404	3072 (Crore)	668	1654	1685	Nucleargreen 2012
Kaiga 3 and 4	2011	404	3282 (Crore)	713	1765	1765	Nucleargreen 2012

The USD-Rupee conversion rate for 2000 through 2011 was approximately 1 USD/46 Rupee (OANDA 2012). The USD is brought to 2011$ using the IHS CERA PCCI (IHS 2012).

The table shows a recent capital cost range of $1000 to $4300 per kWe. The lowest specific cost belongs to Cernavoda Unit 2; however, this unit had begun construction in the 1980s, so its specific cost does not necessarily reflect the costs incurred by greenfield construction. Removing it from consideration, the range for capital costs is then $1700 to $4300 per kWe.

The cost estimates for units under construction range from $1700 per kWe for the Kakrapar and Rajasthan plants (WNA b 2012) to $3468 per kWe for the Atucha plant (Power Technology 2012); this is consistent with the recent costs above.

R5-7. DATA LIMITATIONS

The cost data here reflect construction primarily in non-Western countries, although most of the construction is performed by AECL, a Canadian company.

The cost data here also do not reflect the capital cost of a heavy water enrichment plant. One of India's heavy water plants, Manuguru, had a capital cost of 983 crore in 1992. Based on a USD-Rupee exchange rate of 1 USD/31 Rupee (OANDA 2012), this is 317 million USD in 1992$. This plant has an annual capacity of 185 MT of heavy water. (Ramana b 2007)

The above costs also do not reflect non-fuel operations costs. Since the PHWR is similar to the LWR in many respects, it can be assumed that its non-fuel operational costs are also similar to the LWR costs.

R5-8. COST SUMMARIES

The reference capital cost range for a generic PHWR is assumed to be $1700 to $4300 per kWe in year 2011dollars. This is consistent with the recent historic costs, as well as with the current estimates for plants under construction. However, noting that most of the construction has taken place in non-Western countries, a "Westernization" premium should be applied to account for differences in the regulatory, safety, and industrial practices. Using a 2011 international construction cost survey (Gardiner 2011) as the basis, industrial construction in India ranges from $36 to $63 per square foot, and in China ranges from $50 to $88 per square foot. Comparable construction in metropolitan areas in the United States is greater than $88 per square foot. Assuming US construction would not be in a metropolitan area such as Boston or Los Angeles, that $88 can be used as a proxy value for US construction. The arithmetic mean of the Chinese construction values is $69; the $88 for US construction then represents a 28% premium over the average Chinese. For simplicity, and to reflect great uncertainties in all quantities, this cost estimation will use a 30% premium. Applying a 30% premium changes the estimates to $2200 to $5600 per kWe. These high and low values are shown in the "What-It-Takes" Table R5-3. Previous studies have shown that the PHWR cost per kWe is approximately equal to the LWR cost per kWe (IAEA b); these correspond well to the costs in Module R-1.

Table R5-3. 2012 AFC-CBR Update: Selected "What-It-Takes" Specific Overnight Cost Range (2012 $) for PHWRs.[1]

"What-it-Takes" Specific Cost Range ($/kWe) in year 2011 $	Low	Nominal	High
PHWR	2200	3900	5600
1. For uncertainty analyses a triangular distribution should be used with the values in this table.			

For the 2017 AFC-CBD the 2012$ values for the overnight capital cost are escalated to 2015$ using a factor of 1.088 per the escalation factor table at the beginning of this report .They are then rounded to carry a reasonable number of significant digits. Table R5-4 shows the new values, and Figure R5-6 shows the probability distribution and associated parameters. The O&M cost parameters are the same as for the LWR due to many inherent similarities.

Table R5-4. 2017 AFC-CBR Update: Selected "What-It-Takes" Specific Overnight Cost Range (2017 $) for PHWRs.[1]

What-It-Takes (WIT) Table				
Reference Cost(s) Based on Reference Capacity	Upsides (Low Cost)	Downsides (High Cost)	Expected (Mean Cost)	Selected Values (Mode Cost)
Overnight Cost for NOAK FR in U.S.	$2400/kWe	$6100/kWe	$4200/kWe	$4200/kWe
O&M Fixed Component including D&D fund contribution (no ref. available)	$60/kWe-yr	$87/kWe-yr	$73/kWe-yr	$72/kWe-yr
O&M Variable component including Capital Replacement Component (no ref. available)	0.8 mills/kWh	2.7 mills/kWh	1.8 mills/kWh	2.0 mills/kWh
1. For uncertainty analyses a triangular distribution should be used with the values in this table.				

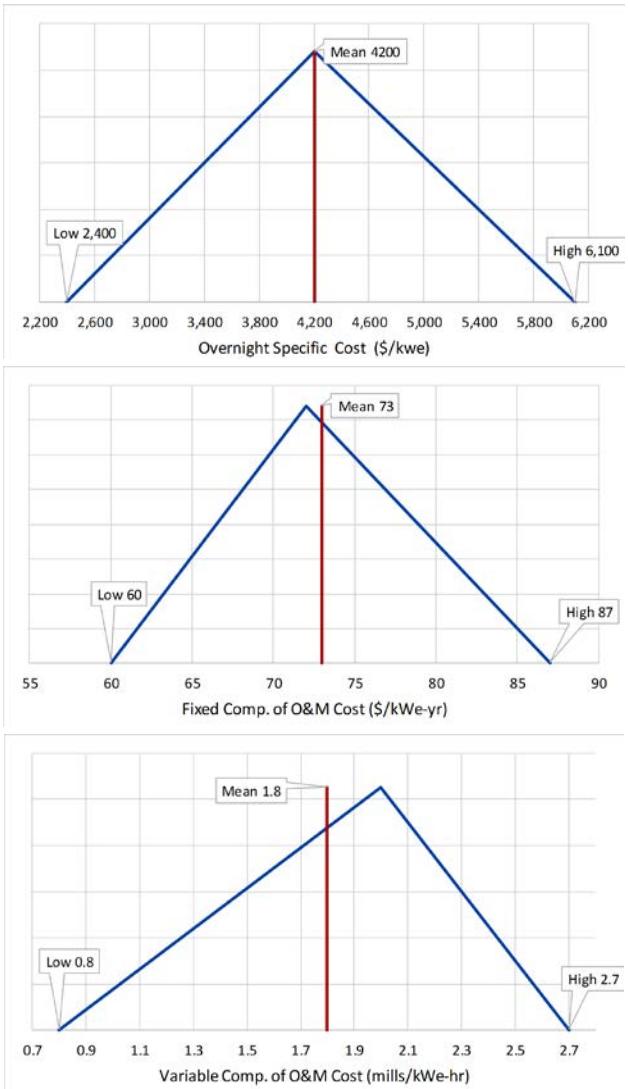

Figure R5-6. Distributions for PHWR Cost Parameters.

R5-9. SENSITIVITY AND UNCERTAINTY ANALYSES

The LUEC for PHWRs would have the same sensitivities to interest rates and construction times as LWRs.

R5-10. REFERENCES

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Module R6

Accelerator-Driven Systems

Module R6

Accelerator-Driven Systems

R6.MD SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2012 AFC-CBR)
- **Estimating Methodology for latest (2012 AFC-CBR) technical update from which this 2017 update was escalated:** Estimate from a few pre-conceptual studies on the concept. No bottom-up estimate was available for a complete accelerator-driven system.

R6.RH REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2012 as Module R6.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2012
- **New technical/cost data which has recently become available and will benefit next revision:**
  - The use of this AFC-CBR information and its format for the Evaluation and Screening Report (INL 2014) did not work mesh well the range of ADS concepts considered during screening. The cost of the accelerator needs to calculated in terms of the size of the accelerator required, which in turn is a function of the degree of sub-criticality and thermal power of the sub-critical reactor. In addition, the electricity requirements (usage) should be included asa lifecycle cost component like O&M. Essentially, instead of rolling all the technology/cost assumptions into a final single $/kWe, the ADS systems analyst should provide them separately so the cost estimator can better account for the specific ADS that they are evaluating.

R6-1. BASIC INFORMATION

This module has been newly drafted for the FY2012 Cost Basis Report update. It is concerned with the capital and operational cost of Accelerator Driven Systems (ADS), defined in this module as industrial scale machines. It is highlighted that no such machine has been constructed nor operated as of yet, therefore all the costs presented here are derived from paper studies. In particular, most of the cost data are derived from the in-depth cost analysis ATW (Accelerator Transmutation of Waste) project (DOE 1999, PNNL 1999) of the late ‘90s. The European MYRRHA (Multi-Purpose Hybrid Research Reactor for High-tech Applications) project is advancing towards demonstration of the technology, with construction scheduled to start in 2015 (INL 2014), but very limited cost data are available as of this writing. While the ATW system was envisioned with the purpose of transmuting transuranics while generating electricity for sale, the MYRRHA machine’s main purpose will be to generate isotopes for research, with a substantially lower power level.

R6-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

ADS are composed of two coupled main parts: (1) a sub-critical reactor in which the power is maintained at the desired level through the use of an external neutron source and (2) a proton accelerator that, through the use of spallation reactions, generates the source neutrons.

It is assumed that, by dropping the constraint of maintaining the nuclear core critical, additional flexibility can be gained by ADS as opposed to critical transmuters[1]. This flexibility in turn can be used for certain types of reactor applications, such as the transmutation of large quantities of heavier MA. In fact, there are not substantial technological differences between fast reactors and the subcritical parts of the ADS (more on this in the rest of this section), to the extent that the cost of FR is assumed as the basis for the cost of the subcritical part of the ADS. Both FR and ADS can achieve similar reduction in the overall TRU/MA inventory and radiotoxicity (OECD/NEA 1993), but the safety characteristics of the two systems are different. Of particular interest are such parameters such as β_{eff} (or the fraction of delayed neutrons emitted in fission), the Doppler coefficient of reactivity and the void effect. β_{eff} in particular is of primary importance to quantify the degree of super-criticality that can be allowed during power excursion: for an ADS loaded with MA only, β_{eff} is about half that of a TRU/Pu burner FR [and this value is in turn about half that of LWR fueled with enriched uranium, from reference (OECD/NEA 1993), page 75]. By being able to operate the ADS in a sub-critical mode, there is a larger margin of safety to the inadvertent/accidental insertion of positive reactivity. The Doppler Effect, another important safety mechanism, is ~1 order of magnitude smaller for MA ADS than for TRU/Pu fast reactors. The values of those important parameters discourage the use of MA-only fast reactors. The subcritical system's degree of subcriticality and the accelerator power level are coupled through the following Equation (7):

$$i_p[\mathrm{A}] = \frac{\nu\left(\frac{1}{\mathrm{keff}}-1\right)}{\varphi^{*}\mathrm{Z}\epsilon_{\mathrm{f}}[\mathrm{MeV}]}\mathrm{P}$$

where

P (in MW) = in a subcritical core of eigenvalue k_{eff},

A = proton current i_p (in A),

ν = the average number of neutrons per fission,

k_{eff} = the standard neutron multiplication number of the subcritical system,

φ^* = the ratio of source neutrons to the average importance of fission neutrons,

ε_f = the energy per fission (in MeV), and

Z = the number of spallation neutrons per incident proton (OECD/NEA 1993).

It is observed that, as k_{eff} becomes smaller, the necessary current becomes larger, increasing the capital and operational cost of the accelerator. Figure R6-1 shows the necessary accelerator power for different sub-criticality levels as a function of the energy of the incident protons (Shropshire et al. 2009). It is observed that no substantial reduction in the accelerator power is obtained for energy higher than 1 GeV, for k_{eff} of 0.95 and 0.98. It is also noted that as reactivity is lost during irradiation, the required beam current (or accelerator power) increases. In turn, reactivity loss is larger for deeper transmutation and longer cycle lengths.

[1]. Fast Reactors are also capable of transmuting large quantities of MA per pass, particularly if designed with a conversion ratio smaller than 1.

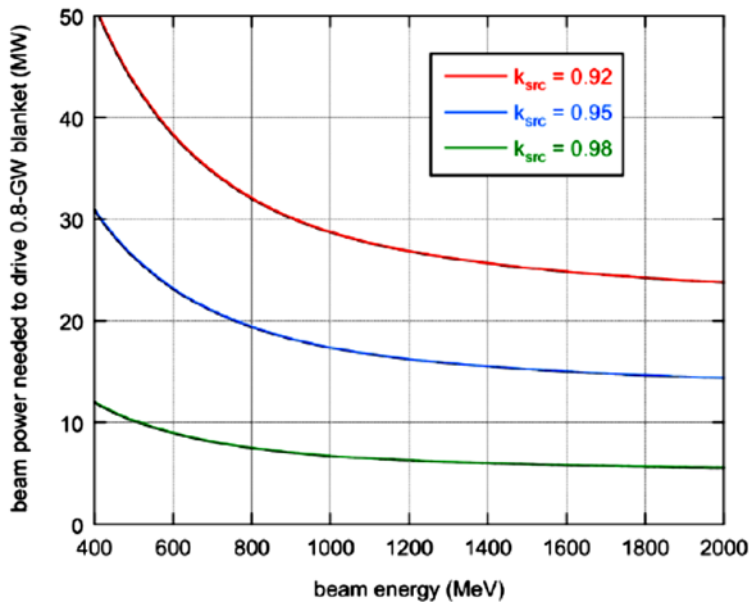

Figure R6-1. Beam power for different accelerator energy, from (Shropshire et al. 2009).

The preferred accelerator of choice for most of the ADS design studies is of the linear type (LINAC): linear accelerators are believed to be capable of generating up to 100 mA of proton accelerated to ~1 GeV (OECD/NEA 1993), so they can potentially be used to generate a continuous beam of up to 100 MW. Cyclotrons are limited to a few MW (probably not more than 10 MW: at present the most powerful is the 590 MeV, 1.8 mA cyclotron at the PSI, Switzerland (De Bruyn et al. 2011), resulting in a power level of ~1 MW), since they are limited by the strength of the magnetic field necessary to keep the ions on a curved path (DOE 1999). LINACS also are capable of nearly continuous particle output: while reactors have a history of reliable/continuous operations, accelerators have a more mixed reliability record (DOE 1999) (i.e. the beam interruption frequency would need to be reduced by 3 orders of magnitude as compared to most operating accelerators (DOE 1999). Even relatively short accelerator shutdown periods can reduce the system availability and degrade the ADS system performance. Additionally, beam interruptions in ADS would be damaging to the subcritical systems due to thermal cycling, shock waves, and loss of power to the grid even if time-integrated availability were to be high. The ATW, and later the AAA (Advanced Accelerator Applications) programs assumed that the causes of failure could be identified and addressed (mostly by improvements and redundancies). The European MYRRHA project (De Bruyn, D. et al. 2011), for a core power of 57 MW_{th}, relies on more proven 1.3 MW LINAC (600 MeV, 2.2 mA) accelerator.

Recommended Procedures for the practical calculations of the capital and O&M cost of an ADS system:

Most types of spallation targets identified in various studies are either (1) the same liquid lead-bismuth (LBE) coolant for LBE-cooled systems (2) tungsten cooled by liquid sodium, for system where sodium is the thermal vector (see Figure R5-4). Some protons are lost at the window between the accelerator and the target, degrading the system performance: to minimize losses, the windows should be thin; however, thick barriers are normally preferred in nuclear system for safety reasons. Some of the European R&D has been devoted towards windowless systems in the past, but the recent effort for the MYRRHA project focused on systems with a window between the beam and the target. Procedures for the practical calculations of the capital cost of an ADS system Procedures for the practical calculations of the capital cost of an ADS system are illustrated below.

The capital cost for ADS system is strongly dependent on the accelerator power: the beam power, P_B, required to generate the neutron source rate necessary to support the subcritical blanket is:

$$P_B[W] = \frac{ns[neutron/s] \times 1.6 \times 10^{-13}[J/MeV] \times E_p[MeV/proton]}{N[neutron/proton]} \tag{5}$$

where E_p is the energy of a proton in the beam, and N is the number of source neutrons generated per proton in the target.

The required accelerator power, P_A, to sustain the required neutron source rate, ns, is easily obtained using the accelerator efficiency, η_A:

$$P_A[W] = \frac{P_B}{\eta_A} = \frac{P_{th} \times E_p \times \nu \times (1 - k_{eff})}{\eta_A \times N \times E_f \times k_{eff}} \tag{6}$$

Equation (6) can be rewritten:

$$P_A[W] = P_{th} \times A \times \left(\frac{1}{k_{eff}} - 1 \right) \tag{6}$$

with $A = \frac{E_p \times \nu}{\eta_A \times N \times E_f}$.

Accelerator efficiencies can be assumed to be of the order of 50%.

Using equation 6, it is possible to calculate the maximum power level of the accelerator, based on the minimum keff during the fuel cycle, i.e. the capital cost should be based on the *maximum* (not *average*) beam power required during the equilibrium cycle. This is the value used to estimate the capital investment needed for this system.

The specific cost of accelerators, normalized per MW of beam power, have a distribution of 35.63 $/W, 180.17 $/W and 360.34 $/W. These costs generally need to be normalized to the power delivered by the system (accelerator plus sub-critical blanket) to the grid, to obtain the overnight cost distributions in ($/kWe). Regarding the subcritical part of the ADS, it is noted that the capital cost in this document are given as normalized per kWe specifically for the ATW system, for which the net electrical efficiency is assumed to be 36.7%. However, for the calculation for the cases with lower electrical efficiency, it is noted that the actual capital investment is more closely related to the amount of thermal power (and thus the dimension) that the system has to be sized for. To obtain a more accurate representation of the likely cost of the system, the specific capital costs from this section should be re-normalized by the ratios of the thermal efficiencies. The specific capital costs are then the sum of that of the sub-critical part of the plant and of the accelerator. For the statistical analysis they must be treated as independent variables and sampled separately, since they have different uncertainty distributions.

R6-3. PICTURES AND DIAGRAMS

Figure R6-2 shows the conceptual scheme of the ATW system. Figure R6-3 shows a Reference ATW target/blanket configuration (Hill et al. 1999). Figure R6-4 shows the main options for spallation targets for the ADS modules as conceived in the ATW project (Hill et al. 1999) and Figure R6-5 from (Hill et al. 1999).

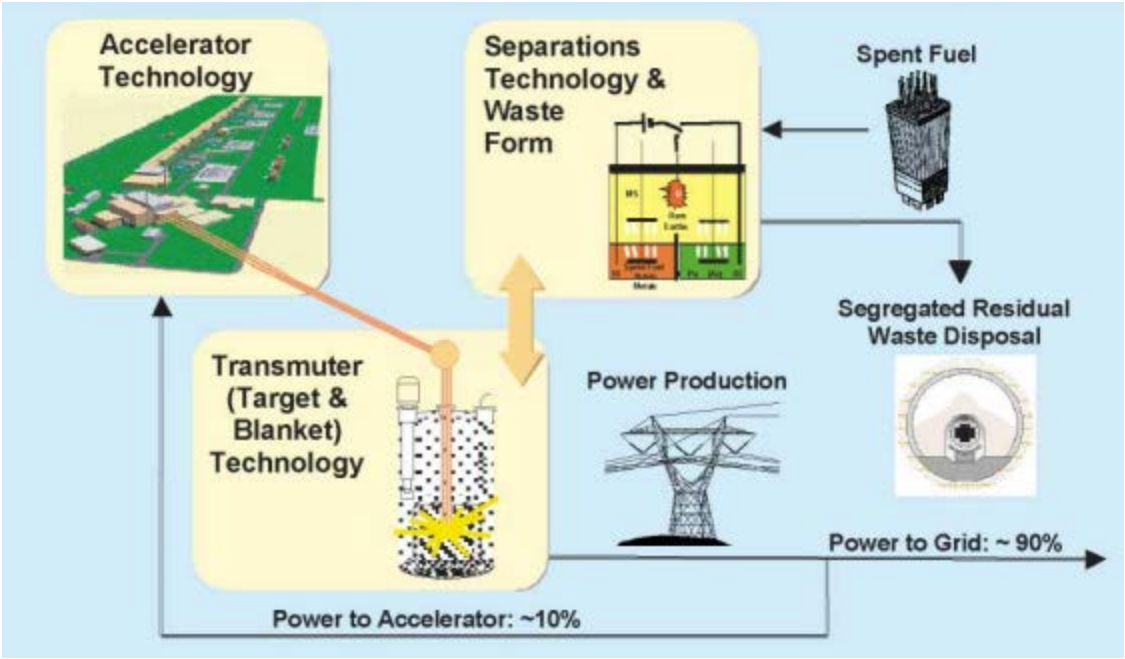

Figure R6-2. Conceptual Scheme for the ATW System (DOE 1999).

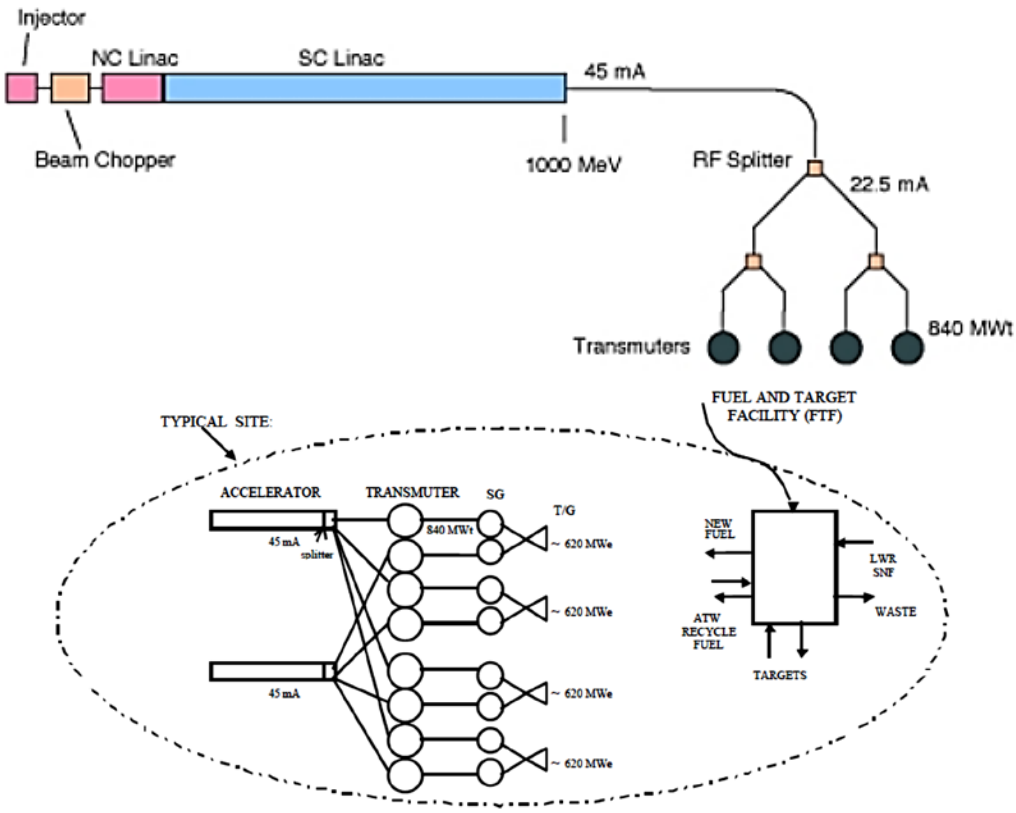

Figure R6-3. Schematic representations of each of the eight envisioned stations of ATW (DOE 1999).

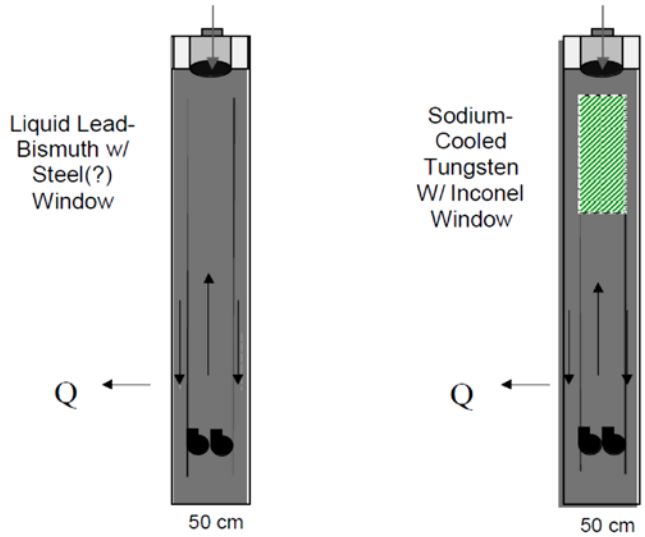

Figure R6-4. Options for spallation target for ADS modules (Hill et al. 1999).

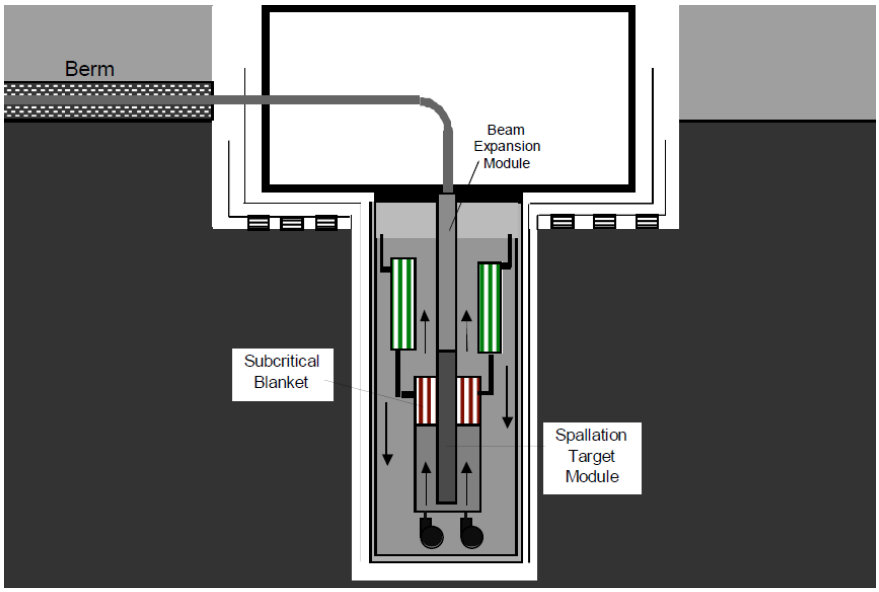

Figure R6-5. Reference ATW target/blanket configuration (Hill et al. 1999).

R6-4. MODULE INTERFACES

The ADS system receives fuel assemblies from the fuel fabrication plant, which can be central or co-located with the ADS facility. After irradiation, the discharged fuel would be kept in wet-storage on site until ready for on-site storage or off-site storage or disposal, or reprocessing. Although both aqueous and electrochemical reprocessing is possible, electro-refining is often envisioned as the fuel reprocessing method for the discharged ADS fuel (DOE 1999, PNNL 1999, OECD/NEA 1993), which would be normally co-located with the ADS. Co-location would save the off-site transportation costs.

R6-5. SCALING CONSIDERATIONS

While the subcritical system will feature a reduction in specific costs as the system power increases, the same cannot be said for the large accelerators required for ADS systems, due to the complexity and uniqueness of large accelerators with low beam trip required for these kinds of applications.

R6-6. COST BASES, ASSUMPTION AND DATA SOURCES

The ATW cost basis is used as a reference cost assessment, mostly because of the detailed economic study performed for the ATW roadmap. The following ATW systems was envisioned (DOE 1999, PNNL 1999, Hill et al. 1999) to operate for a period of 117 years: 8 stations: each station has 2 accelerators feeding 4 power blocks, in turn containing 2 sub-critical reactors and 1 turbine, for a total of 64 sub-critical reactors, generating 2480 net MW_e per station (see Figure R5-3). The cost summary is divided in two parts: (1) the accelerator (capital and O&M) and (2) the subcritical reactor and the power conversion equipment (capital and O&M). In this section, the fuel processing/fabrication facilities are excluded, since they are treated in detail in Section D2/F2.

Capital and O&M Cost of the Subcritical Reactor

The basic assumption in the ATW cost study is that the capital cost of the sub-critical part of the system will be similar – in fact, slightly higher – to that of a critical fast reactor similar in size/power level to the sub-critical unit. The specific cost (i.e. $\$/kW_e$) of the reactor/power conversion is going to be higher than that of a similar critical fast reactor because (PNNL 1999):

1. The extra size of the plant necessary to generate the electricity needed to run the accelerator: this is electricity that is not available for sale. The extra electricity needed is about 8% of the total in the case of ATW: each subcritical unit has an electric output of 310 MW_e, and needs an accelerator with a power level of 11.25 MW. Considering that the typical accelerator efficiency is 45%, the electric consumption of the accelerator is about 25 MW, or about 8% of the total.
2. In addition to the standard FR components, there will be extra complications such as target and accelerator/target connections.
3. Some components will be absent or reduced, such as control rods, but the cost benefit of this is likely to be over-compensated by the extra cost of components needed in ADS and not in FR (PNNL 1999).

The Advanced Liquid Metal Reactor (ALMR) has been used as a reasonable cost basis for ATW in (PNNL 1999), because of the large amount of work done on the cost of the ALMR (funded by DOE from 1989 to 1995 (DOE 1999)). Table R6-1 gives the capital cost for the ATW subcritical reactor and for the steam generator and turbine, from (PNNL 1999) data converted from 1999 $ to 2009 $ using a CPI deflator of 1.29 (Williamson 2010), both for the First of a Kind (FOAK) and for the Nth of a kind (NOAK). Because of the additional complexity outlined above, the capital cost of the subcritical unit and of the site support are increased from the ALMR values by about 10% (or ~70m$ per unit) according to judgments made by the experts of the ATW group (PNNL 1999). The turbine/generator section of the plant is left un-affected, since it is reasonable to assume that it will be the same (thus having the same costs) for critical and sub-critical units. The specific cost in $\$/W_e$ are to be normalized by the net electricity available for sale, equal to the total electricity produced by the subcritical unit minus the electricity consumed to run the accelerator.

Table R6-1. ATW capital cost (from (DOE 1999), converted from 1999 % to 2009 % using the CPI deflator of 1.29).

Subcritical Part of ATW	310 MW_e Reactor Capital Cost (m$)	Reactor Specific Cost ($\$/W_e$)	620 MW_e Turbine Cost (m$)	Turbine Specific Cost ($\$/W_e$)	Site Support Cost (m$)	Power (MW_e) Supported by Site Support	Site Support Specific Cost ($\$/W_e$)	Total Specific Cost ($\$/W_e$ to the grid)
FOAK	954.21	3.35	505	0.89	225	620.00	0.40	4.63
NOAK	627.84	2.20	389	0.68	210	2480.00	0.09	2.98

The O&M costs for the sub-critical part and for the power conversion of the power station of the ATW system are shown in Table R6-2 in 2009 $, converted using the CPI deflator (Williamson 2010). The values are based on the FOAK values reported in (PNNL 1999), for the ATW studies adapted from

the ALMR data. The NOAK data were derived by the authors by assuming that the O&M cost (in million$/y) will remain the same from the 1st to the Nth unit, with the difference that the power level will increase. For this reason, the overall specific O&M yearly cost will decrease from 229 to 131 $/(kW$_e$y)[2].

Table R6-2. ATW O&M cost (from (PNNL 1999), converted in 2009 $).

Subcriztical Part of ATW	310 MW$_e$ Reactor O&M Cost (m$/y)	Reactor Specific Cost ($/W$_e$ y)	620 MW$_e$ Turbine O&M Cost (m$/y)	Turbine Specific O&M Cost ($/W$_e$ y)	Site Support O&M Cost (m$/y)	Power (MW$_e$) Supported by Site Support	Site Support Specific O&M Cost ($/W$_e$ y)	Total O&M Cost Specific to the Net Electrical Output ($/kW$_e$ y)
FOAK	21.54	0.08	13.16	0.02	74.30	620.00	0.13	229.03
NOAK	21.54	0.08	13.16	0.02	74.30	2480.00	0.03	131.26

Capital and O&M Cost of the Accelerator Part

The accelerator cost data are available from the APT project (Accelerator Production of Tritium) (PNNL 1999), where the estimates have been derived within 10 years of collaborative work between two industrial partners, General Atomics and Burns & Roe Enterprises. The reference accelerator for the APT project is a LINAC with a 1 GeV, 100 mA proton beam, for an accelerator power of 100 MW. The accelerator costs from the APT program – adapted by the expert group for the ATW project – are summarized in Table R6-3, where the values for FOAK and NOAK are both reported, in 2009 $ adjusted using the CPI deflator, for the reference linear 1 GeV accelerator. The FOAK accelerator has 12 MW of output, and will feature a substantially higher design and construction costs than the NOAK, resulting in a specific capital cost of 180 $/W. The NOAK is expected to feature a substantially lower capital cost for design and construction, resulting in a specific capital cost of 35.6 $/W. When these costs values are normalized to the electric output available for sale (i.e. output of the turbine/generator of the subcritical multiplier minus the electricity consumed by the accelerator itself, considering the fact that the typical accelerator efficiency is 45%), the specific capital costs in 2009 $ for the FOAK and NOAK accelerator are 7110 $/kW$_e$ and 1410 $/kW$_e$ respectively.

Table R6-3. Accelerator total and normalized power costs (from (DOE 1999), converted in 2009 $ using the CPI deflator).

Linear Accelerator	Design Cost (m$)	Construction Cost (m$)	Current (mA)	Total Capital Cost (m$)	Capital Cost Specific to the Accelerator Power ($/W)	Accelerator Power (MW)[1]	Cost of Accelerator/ Subcritical Unit (m$)	Specific Accelerator Capital Cost ($/W$_e$ to the grid)
FOAK	548.25	1613.79	12	2162.04	180.17	11.25	2026.91	7.11
NOAK	211.56	1391.91	45	1603.47	35.63	11.25	400.87	1.41
1. This is the power necessary to drive each 310 MW$_e$ subcritical unit								

The O&M of the accelerator part of the ATW are shown in Table R6-4, in 2009 $, in absolute value (i.e. million$/y) and in normalized value ($/kW$_e$ y).

Table R6-4. Accelerator O&M cost (from (PNNL 1999), converted in 2009 $ using the CPI deflator).

Accelerator	Current (mA)	O&M Cost (m$/y)	Specific O&M Cost ($/W y)	MW/Each 310 MW Subcritical Unit	O&M Cost of Accel/Unit (m$/y)	Accelerator O&M Cost $/kW$_e$ y
FOAK	12	78.69	6.56	11.25	73.77	258.85
NOAK	45	56.76	1.26	11.25	14.19	49.79

[2]. This value is to be used to easily obtain the annual O&M cost by multiplying the net electrical output of the system by the value in the table: for example, for a 10^6 kW$_e$ system, the annual O&M cost of the NOAK subcritical unit would be 131.26 million $.

R6-7. DATA LIMITATIONS

No ADS has been constructed and operated to date, therefore the cost assumptions presented here are largely estimates of costs based on paper studies. In fact, most of the data in this revision rely on a single cost study (the ATW effort of the late '90s) that contain the most detailed and complete effort to estimate not only the R&D costs, but also the cost of a NOAK system.

While few critical fast reactors have been constructed around the world – therefore providing both a demonstration of the technical feasibility and some base for FOAK cost estimates – no accelerator has been built of the power level required to drive an industrial-scale ADS. An increase in the power level by 1-3 orders of magnitude as compared to the currently most powerful machines, appears to require a technological leap. Additionally, a reduction in the beam interruption frequency by 3 orders of magnitude as compared to the present accelerator's performance, seems to also require a technological leap. As an example of the excessive optimism that may be contained in the ATW cost data, it is noted that, by private conversation of the authors with the designers of the European MYRRHA facility, it was possible to obtain a specific cost of that machine in the order of 200-300 $/W, higher than the FOAK specific cost proposed for the ATW project of 180 $/W, while the accelerator for the MYRRHA project features a power level within the limits of available technology. The spallation target technology also appears not fully demonstrated, as well the connection/interface between the accelerator and the subcritical reactor. The combined effect may affect the technological feasibility of such a system, and will have an impact on cost which is difficult to predict at this early stages of technological maturity, not to mention the extrapolation to the cost of an NOAK facility.

R6-8. COST SUMMARIES

The specific costs of the ATW system, as representative of an ADS system (in 2009 $), are summarized in Table R6-5. It is observed that there is a large variation in specific cost between the FOAK and the NOAK construction costs, mostly attributable to the large cost variation of the accelerator. To justify this large reduction in accelerator costs, the learning curve for the accelerators was set at 85%, while that of the nuclear systems was assumed at 95%. The capital costs reported here do not include decommissioning and decontamination costs, which are assumed as 10% of fabrication cost for activated parts (Murphy 1984), and at 5% for non-contaminated parts such as some of the accelerator's components. Decommissioning costs can be assumed to follow the same 95% learning curve as the nuclear system.

Table R6-5. Summary of the specific capital cost of the ATW system, in 2009 % as representative of an ADS system.

	Subcritical Reactor + Power Conversion Cost ($/kW$_e$)	Accelerator Cost ($/kW$_e$)	Total Specific Cost ($/kW$_e$)	Accelerator Cost as a % of Total Cost
FOAK	4630	7110	11740	60.6%
NOAK	2980	1410	4390	32.1%

The total normalized O&M of the whole ATW system in 2009 $ is shown in Table R6-6.

Table R6-6. Summary of the specific O&M of the ATW system, in 2009 $, as representative of an ADS system.

	Subcritical Reactor + Power Conversion O&M Cost ($/kWey)	Accelerator O&M Cost ($/kWey)	Total Specific O&M Cost ($/kWey)	Accelerator Cost as a % of Total Cost
FOAK	229.03	258.85	487.88	53.1%
NOAK	131.26	49.79	181.05	27.5%

The operation and capital costs of ADS are higher than those of critical reactors, mostly because of the added costs of the accelerators. However, even if the accelerator capital cost were 0 and the subcritical reactor would cost the same to build and operate as a critical reactor, ADS would still be more expensive than FR because of the relatively large fraction of the electricity produced that is needed to run the accelerator (about 8% in the ATW study) and consequently would not be available to generate revenue.

No reliable uncertainty ranges could be obtained to-date on the O&M and capital costs of ADS. However, the higher specific costs of the MYRRHA accelerator, for example, would suggest that a cost of the FOAK facility could be twice as high as the specific value suggested in the ATW work, or as much as 300 $/W. It is further noticed that the degree of learning implied in the accelerator cost reduction from FOAK to NOAK may never materialize. It therefore appears prudent to the authors, considering the degree of technical immaturity of this technology, to assign the NOAK ATW specific cost as the lowest, most optimistic scenario, the FOAK ATW as nominal value and twice that value consistent with the MYRRHA accelerator suggested specific cost – as a high (or most pessimistic) value. Table R6-7 summarizes the "what-it-takes" values for the specific overnight capital and O&M costs for both the accelerator and the sub-critical parts.

For the NOAK case, the sub-critical reactor part of the capital cost has been estimated at 2980 $/kW$_e$, by scaling up this value from the ALMR cost of 2350 $/kW$_e$. If these costs were higher, as for example suggested in [Shropshire et al. 2009] (i.e. 4200 $/kW$_e$), the specific cost of the subcritical part would be correspondingly higher. The ATW estimated NOAK cost has been therefore adopted as lower boundary (or most optimistic scenario), and the values of [Shropshire et al. 2009], scaled up by 10%, as selected and upside values.

Table R6-7. What-It-Takes Cost Summary Table.(2012$)

	Upside (Low Cost)	Selected Value (Mode Cost)	Downside (High Cost)
Capital Cost of the Subcritical Reactor	2980 ($/kW$_e$)	4620 ($/KW$_e$)	7700 ($/kW$_e$)
Capital Cost of the Accelerator	1400 ($/kW$_e$)	7100 ($/KW$_e$)	14200 ($/kW$_e$)
O&M Cost of the Subcritical Reactor	60 ($/kW$_e$y)	131 ($/KW$_e$)	230 ($/kW$_e$y)
O&M Cost of the Accelerator	50 ($/kW$_e$y)	153 ($/KW$_e$)	256 ($/kW$_e$y)

Table R6-8. Shows the same Table with all cost numbers escalated to Year 2017 dollars:

Table R6-8. What-It-Takes Cost Summary Table (2017$)

	Low Cost	High Cost	Mean Cost	Mode Cost
Capital Cost of the Subcritical Reactor	3200 ($/kW$_e$)	8400 ($/kW$_e$)	5500 ($/KW$_e$)	5000 ($/KW$_e$)
Capital Cost of the Accelerator	1500 ($/kW$_e$)	15400 ($/kW$_e$)	8200 ($/KW$_e$)	7700 ($/KW$_e$)
O&M Cost of the Subcritical Reactor	65 ($/kW$_e$y)	250 ($/kW$_e$y)	153 ($/KW$_e$)	143 ($/KW$_e$)
O&M Cost of the Accelerator	54 ($/kW$_e$y)	278 ($/kW$_e$y)	166 ($/KW$_e$)	166 ($/KW$_e$)

Regarding the O&M costs, it is noted that the FOAK and NOAK values suggested for the ATW appear to have substantially higher values and span a substantially larger range than the values suggested in [Shropshire et al. 2009] for the fast reactors. Therefore it is retained prudent to adopt the low fast reactor value of reference [Shropshire et al. 2009] as Upside (Low Cost), the NOAK FOAK ATW O&M costs as selected and downside values, respectively. For the accelerator, the NOAK and FOAK O&M costs have been adopted as Low and High estimates, respectively. Figure R6-7 shows the probability distributions for the specific capital cost and specific O&M cost for the two parts of an ADS system.

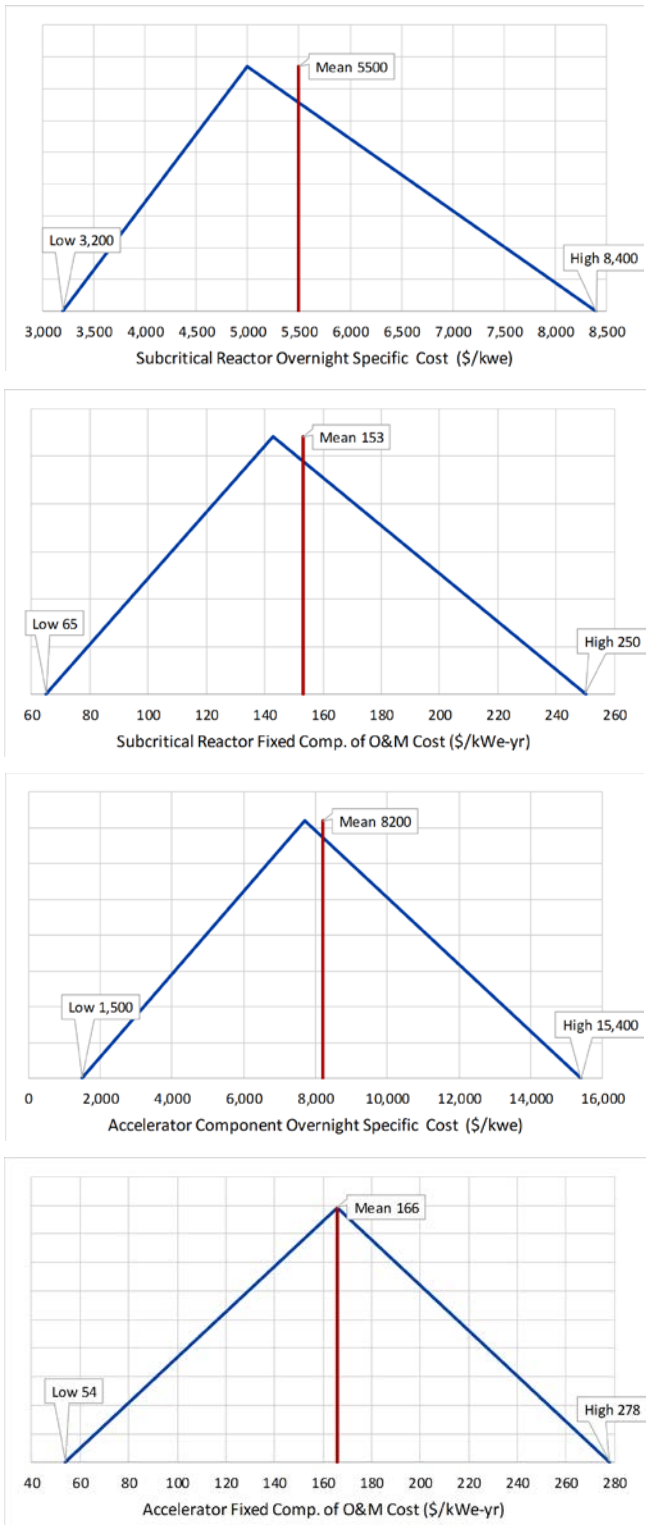

Figure R6-7. Frequency distributions and parameters for the cost elements of a generic ADS system.

R6-9. SENSITIVITY AND UNCERTAINTY ANALYSES

None available.

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Module R7

Liquid-Fueled Salt-Cooled Reactors

Module R7

Liquid-Fueled Salt-Cooled Reactors

R7-MD. SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2012 AFC-CBR)
- **Estimating Methodology for latest (2012 AFC-CBR) technical update from which this 2017 update was escalated:** Some bottom-up MSR design and cost estimates from ORNL work in the 1970s were utilized to develop updated cost estimates combined with engineering judgment of the MSR's differences from LWRs.

R7-RH. REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2012 as Module R7.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2012
- **New technical/cost data which has recently become available and will benefit next revision:**
  - A large interest in this type of reactor has evolved recently. This recent attention to MSR concepts is mostly by private companies, which are likely to keep their cost estimates proprietary. There is likely to be some new MSR economics information available, but the authors are unaware of any specific well-documented information. A recent report (EIRP 2017) on advanced reactor economics used some of this private industry post projection data ($/kWe for various NOAK designs) to examine overall economic feasibility of advanced reactors (of all types) as a class vis-à-vis conventional reactors.

R7-1. BASIC INFORMATION

(Module R7 was a new addition to the 2012 AFC-CBD Update report). Liquid-fueled Salt-cooled reactors, more **commonly referred to in the literature as just Molten Salt Reactors (MSRs)** represent a class of reactors that involve the use of dissolved fuel in fluoride or chloride salts that also serve as the coolant material. (It should be noted that a solid-fueled, high temperature reactor can have a molten-salt coolant. This type of Advanced High-temperature Reactor (AHTR) is the subject of Module R8.) The fuel-containing salt can additionally be processed, either online or in a batch mode, to allow the removal of fission products and the introduction of fissile fuel and fertile materials during the operation of the reactor. MSRs concepts have been developed with both thermal and fast neutron spectrums and with uranium, thorium, and plutonium fuels. The MSR is most commonly associated with the U-233/Thorium fuel cycle as the nuclear properties of U-233 combined with the online removal of parasitic absorbers results in the ability to design a thermal-spectrum breeder reactor.

An extensive program supporting research and development of a thermal-spectrum Molten Salt Breeder Reactor (MSBR) at ORNL in the 1950s – 1970s resulted in reactor designs and a significant amount of technology development in materials, salt technology, and reactor components. This research program included the operation of the Molten Salt Reactor Experiment (MSRE) in the late 1960s. As a result of rising proliferation concerns an alternative design was developed in the 1970s, the Denatured Molten Salt Reactor (DMSR) concept was developed in which there is minimal fuel salt processing and

uranium is added to ensure that there is sufficient U-238 present to “denature” the U-233 and U-235 (a non-proliferation objective). The molten salt reactor has been selected as one of the six Generator IV system concepts. China is now pursuing this option in earnest with a $350M R&D program. This is in part driven by the fact that China has considerable thorium reserves. Alternative concepts have been proposed with a fast neutron spectrum [Renault2009] and to support actinide-burning applications [Ignatiev2005]. In addition, molten salts have also been proposed in the use of Accelerator Driven Systems (ADS) [Bowman 1998] and Fission-Fusion Hybrids (FFH) [Lee 1981]. This is the first time the MSR has been considered in the AFC-CBR, hence more technical descriptions of the reactor and fuel handling system are present in this module.

The MSBR provides an example of a full recycle system in which the nuclear fuel is fully recycled with only fission products and other processing wastes being disposed. The DMSR could be considered in the once-through or modified open cycles based on the level of fuel processing that is performed. At the time of development, ORNL considered the system a once-through system but with definition given above, the DMSR is an example of a modified open system that involves minimal processing of the fuel, improved source material (U or Th) resource utilization (in comparison to LWRs) and with disposal of fuel as well as fission products. These systems were selected as illustrative examples for this study primarily because of the availability of a significant amount of historical design information and the relatively complete concept design documents that are available. Alternative concepts, as discussed above, can be included in future assessments based on the available and development of technical information.

Since considerable economic analysis data were available from early and recent (Spring 2010) ORNL documents, it has been decided to include the details of the latter unpublished life cycle cost estimate for the thermal MSR, which was prepared by the author of this Module R7 section shortly before he retired from ORNL, as part of this document. This economic assessment was undertaken as part of series of MSR assessments undertaken by ORNL staff as part of the early Fuel Cycle Options Assessment Program funded by NE-FCRD. Some technical details prepared by other ORNL staff are included and form the design basis for the cost estimate.

R7-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

Basic Reactor Features. The unique characteristic of Molten Salt Reactors is the use of liquid rather than solid fuel. The use of a liquid fuel allows many reactor design features that are not possible with solid fuel. These include circulation of the fuel-containing liquid to act as a coolant and heat transfer mechanism, performing on-line chemical processing to remove parasitic absorbers and optimize breeding and burning of materials, and a different means of passive safety, such as draining the fuel from the core. The MSR designs of the 1960s and 1970s were focused on optimizing the thorium cycle to achieve a high level of breeding performance by on-line chemical processing. At the time, it was envisioned that there would be a very quick growth in nuclear energy with fissile material availability representing a limit to growth. The fuel for MSRs consists of fissile and fertile actinides dissolved in a liquid carrier salt. The most common salt that is used is a LiF-BeF2 salt with lithium being enriched in its Li-7 content to minimize neutron absorption and tritium production. Many other salts have been considered based on sodium, zirconium, rubidium and other materials. In addition for fast-spectrum systems, chloride-based salts have been considered, however, the majority of all research and development up to 2012 has been with fluoride salts, which will be the primary focus in this module.

The on-line chemical processing system is fundamentally based on fluoride chemistry to allow effective removal of the uranium from the salt, followed by vacuum distillation for the removal of fission products (with the gaseous fission products being easily removed by helium sparging). The removal of the highly-absorbing fission products as well as allowing the U-233 precursor, Pa-233, to decay outside of the core, results in an optimal breeding system for a thorium reactor (breeding ratio ~1.07). Combined with the low fissile loading, the thermal spectrum MSR can achieve doubling times that are similar to fast

spectrum systems with a much higher breeding ratio. The core region of the reactor consists of a matrix of graphite blocks that provide moderation to create a critical system with interspersed fuel and fertile blanket regions for a two-fluid reactor.

The reactor operates at a high temperature, with the fuel salt core exit temperature of 700 °C and was originally a design based on a steam power conversion system with a 44% power conversion efficiency. Current concepts would likely utilize a Brayton cycle, which is a better match with the high temperatures of the reactor. The safety of the reactor was ensured by its negative reactivity coefficients and the use of freeze plugs with a drain tank system with passive decay heat removal. Should the fuel salt temperature increase, the freeze plugs will melt and the fuel will drain into tanks that are in a subcritical configuration and have sufficient decay heat removal. Given that the fuel salt will distribute radioactive materials throughout the primary fuel circuit, the system was designed for remote maintenance, which was demonstrated in the operation of the MSRE.

Fuel Cycle Application of MSRs. In the traditional thorium-based breeder application, the MSBR provides a long-term option for nuclear energy based on the large quantity of available thorium. Additionally, the waste stream is comprised primarily of fission products as the higher actinide production is relatively low in the thorium-uranium cycle and the actinides are circulated in the fuel salt until fissioned. While first conceived of as a breeder reactor, the MSR concept was further extended at ORNL to a Molten Salt Converter Reactor (MSCR) when it became apparent that fuel resource availability (U and Th) would not be a concern for a considerable time. The MSCR differed from the MSBR in that it had a simpler chemical processing system since it did not have to achieve a conversion ratio greater than unity. Additionally, as concerns with proliferation of reprocessing technology (DOE 1980) increased in the late 1970s, a Denatured Molten Salt Reactor (DMSR) was developed without on-line processing and with the addition of low-enriched uranium to ensure that the fissile uranium content was in the LEU range. Of course, the addition of low-enriched uranium will increase the higher actinide production over that of the pure U-233/thorium system.

In the past decade, interests in MSRs for the mission of actinide management have become more prominent in France, Czech Republic, and Russia. Several concepts have been developed to use minor actinides and transuranics as fuels with the waste products being predominately fission products. The Czech Republic concept is known as SPHINX and consists of a fast-spectrum MSR with fuel based on plutonium and minor actinides from used LWR fuel. [Hron 2009] The French have studied a thermal-spectrum burner (AMSTER) that uses spent LWR tranuranics as fuel with thorium support. [Vergnes 2001] In Russia a fast-spectrum molten salt reactor transmuter concept has been developed (MOSART). [Ignatiev 2005] More recently, a molten salt fast reactor (MSFR) has been developed in France [Delpech 2009]. China now has a $350M RD&D program to develop this technology and India is still pursuing it; however, not necessarily for actinide management. A number of new concepts are now (2017) under development by companies such as Terrestrial Energy (Canada), Moltex Energy (UK), ThorCon Power (USA), Flibe Energy (USA), Transatomic Power (USA), and TerraPower (USA). At the time of writing this Module R-7, no data was available to evaluate any of these concepts.

R7-3. PICTURES, DIAGRAMS, AND DEPLOYMENT STATUS

A large number of systems configurations based on MSR are possible and many have been considered to a varying degree of detail. Based on the availability of information two concepts have been chosen that provide representative systems for both full recycle and modified open fuel cycle options. These concepts include:

- Molten Salt Breeder Reactor – full recycle system concept based on U-233/Thorium fuel cycle
- Denatured Molten Salt Reactor – modified open fuel cycle system based on thorium/LEU fuel cycle with limited processing.

In addition to these concepts, an additional system based on an actinide burning system (light water reactor used fuel into a molten salt reactor) could additionally be considered as an additional modified open fuel cycle system. This Module considers only the MSBR.

Molten Salt Breeder Reactor (MSBR) Description/Schematic. The MSBR concept selected for this work is that of the final ORNL design, which was based on a single fluid system [Robertson 1971 and Bettis 1970]. A conceptual layout of the reactor is shown in R7.1. The design is a single-fluid concept that contains ~ 43 m^3 of fuel salt (71 mole% ^7LiF, 16 mole% BeF_2, 12 mole% ThF_4, and ~0.3 mole% $^{233}UF_4$) as per Figure R7-2. The plant is a four-loop design with an average core power density of approximately 22 kW/liter. A total of 295,000 kg of graphite was used in the design and the approximately 205,000 kg of that was to be replaced approximately every 4 years A summary of the MSBR key design and operating parameters is presented in Table R7-1.

Table R7-1. MSBR key design and operating parameters.

Parameter	Value
Reactor Thermal Power (MW)	2250
Reactor Electrical Power (MWe)	1000
Fissile fuel inventory (kg)	1501
Thorium inventory (kg)	68,100
Thorium feed rate (kg/yr)	~6000
Inventory U/Np/Pu/Am/Cm (kg)	1988/15.3/13.4/2.3/6.2
Waste Th/Np/Pu/Am/Cm (kg/GWe-yr)	5400/0.72/0.63/0.11/0.29
Waste Total TRU (kg/GWe-yr)	1.74
Breeding ratio	1.06
Doubling time (years)	22
Fuel salt components	^7LiF-BeF_2-ThF_4-UF_4
Fuel salt composition (mol %)	71.7-16-12-0.3
Core inlet/outlet temperature (°C)	566/704 °C

In terms of core design parameters the MSBR core, shown in Figure R7-1, has a peak power density of 70.4 kW/liter and an average power density of 22.2 kW/l. Graphite in the reflector region was expected to last the 30-year life of the reactor. The maximum flow velocity in the core was estimated to be 2.6 m/s.

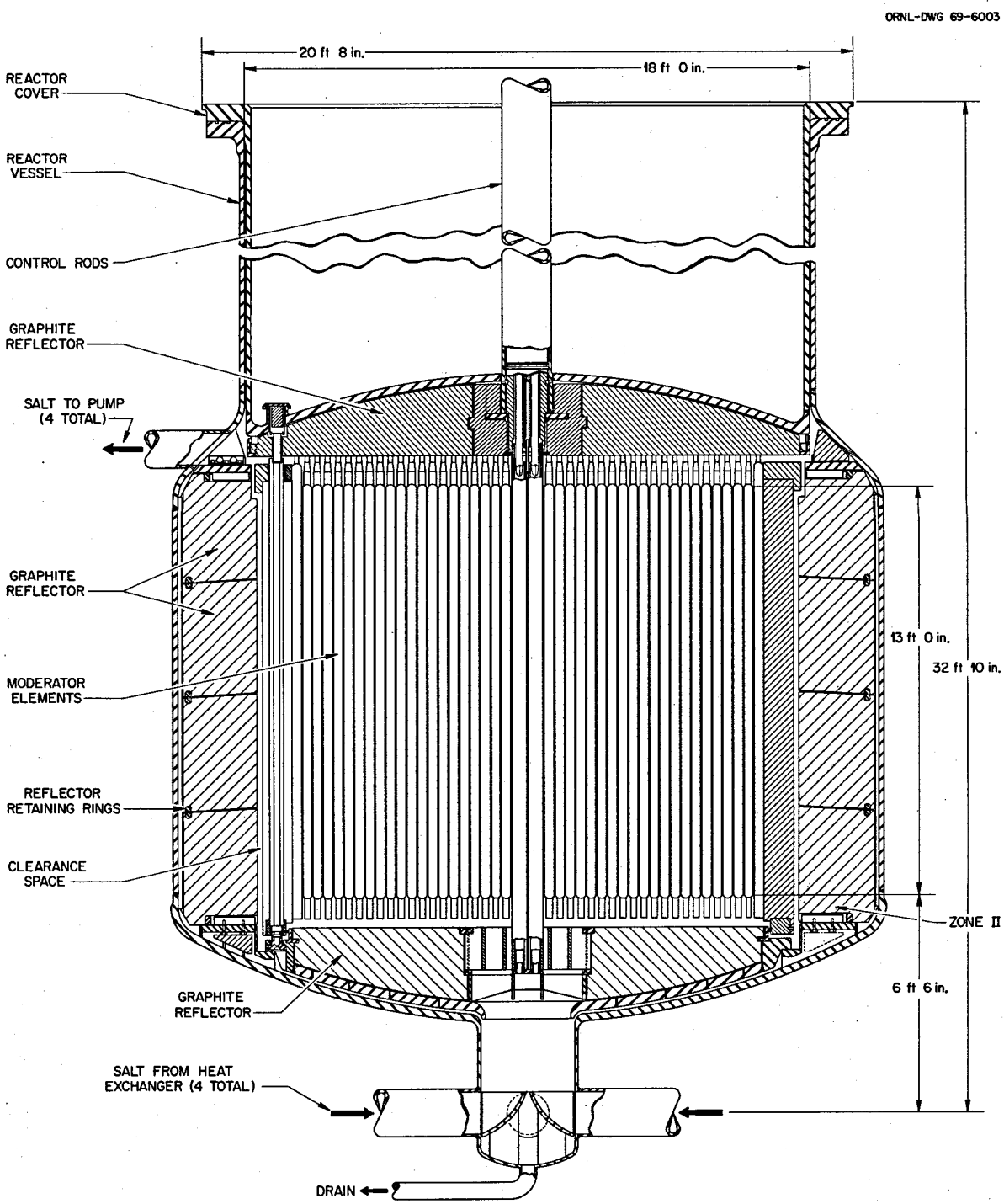

Figure R7-1. Single fluid MSBR core diagram [Robertson 1971].

The total volume of the fuel salt was estimated to be 48.7 m^3 with approximately 30.4 m^3 of fuel salt in the core. The continuous salt-processing system, a schematic of which is shown in Figure R7-2 consists of two steps: (1) the removal of uranium and protactinium from salt leaving the reactor along with reintroduction of uranium and (2) removal of rare-earth fission products. The flow rate of the processing stream is approximately 3.3 liters/min. In the first process a fluorinator removes approximately 95% of

the uranium as gaseous UF_6. The salt then flows to a reductive extraction column where protactinium and the remaining uranium are chemically reduced and extracted into liquid bismuth in a counterflow arrangement. The bismuth contains lithium and thorium as reducing agents that are added at the top of the extraction column. It is not clear whether the latter extraction process was demonstrated on anything other than a small scale, but the initial fluorination process and uranium reintroduction was demonstrated during MSRE operation.

The TRU inventory is very low in this system, with the Pu inventory being ~15 kg and the total TRU inventory being ~37 kg. This will result in a very low discharge fuel TRU content of actinide wastes.

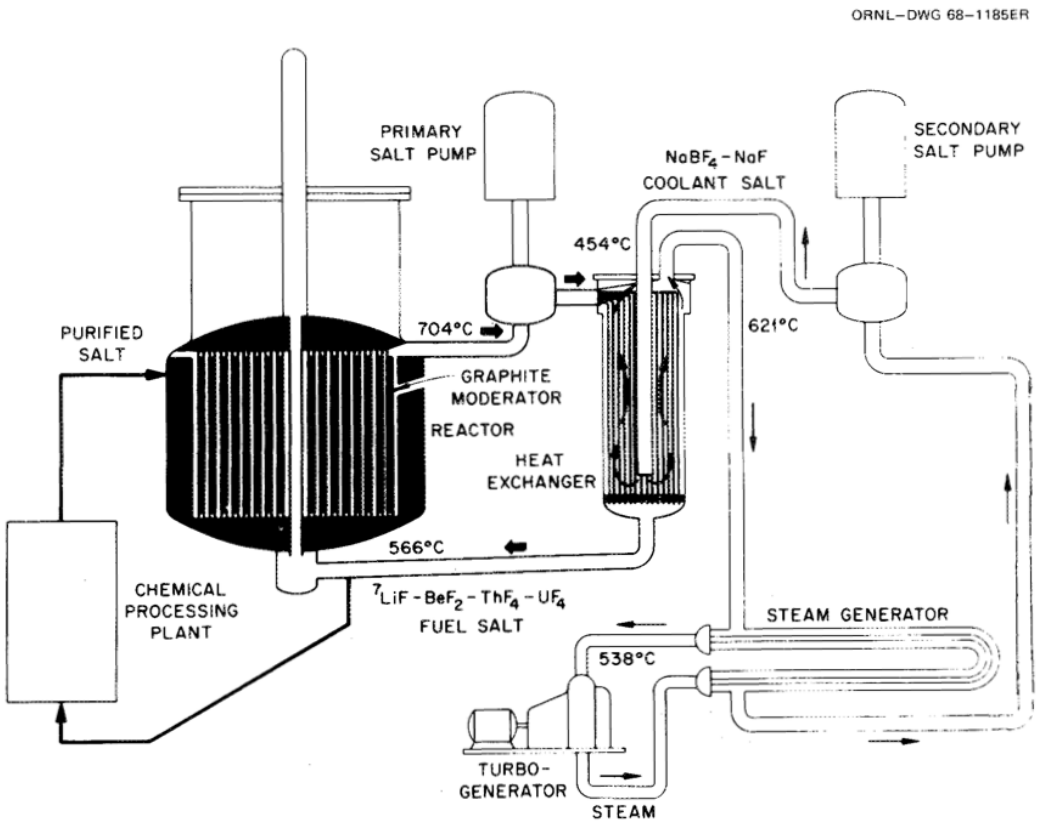

Figure R7-2. Single fluid MSBR key system diagram [Engel 1980].

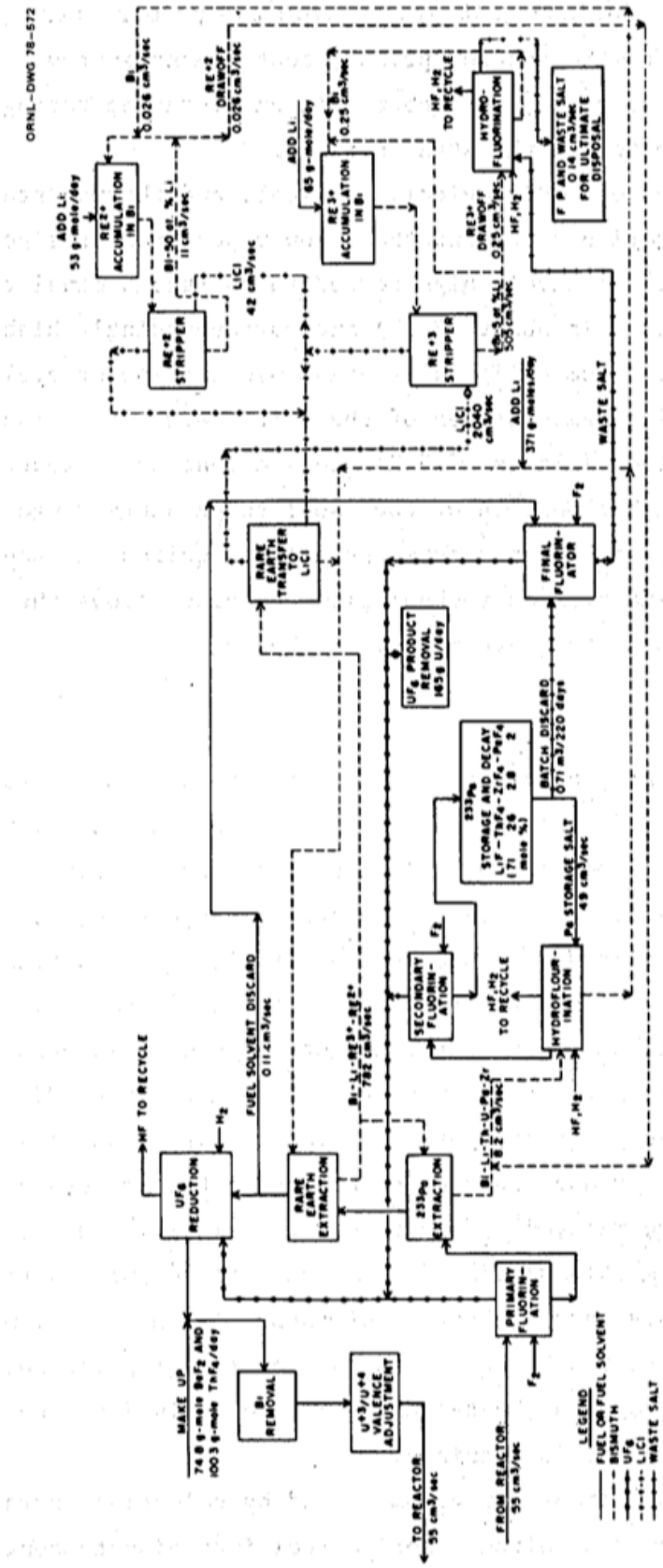

Figure R7.3. Schematic of a single-fluid reactor concept salt processing flow [Engel 1978].

R7-4. MODULES INTERFACES

This reactor concept is unique by the fact that the fuel is part of the coolant. The fabrication of solid fuel forms required by most reactor types is not required. It is likely that the heavy-metal containing chemical compounds would be added as powders for dissolution in the salt. Uranium enrichment requirements would depend on the type of MSR deployed and whether enriched uranium is required as a startup material. Some conversion steps for both uranium and thorium would be required to prepare the correct chemical compound (salt) for use as part of the coolant.

The fuel cycle back-end will need to process the waste materials withdrawn continuously from the molten salt coolant. Little detail is available on the treatment and packaging processes required. These processes are likely to be variants of those described in the "G"-modules.

R7-5. SCALING CONSIDERATIONS

Since this reactor concept is in large part a chemical plant, scaling rules for chemical plants are likely to apply.

R7-6. COST BASES, ASSUMPTIONS, AND DATA SOURCES

The basic objective for a reactor concept is to "maintain an economical nuclear fuel cycle." The corresponding parameters for fuel cycle economic evaluation are as follows:

- Overnight cost for the reactor (including the $/kWe "specific" cost)
- Recurring operations and maintenance (O&M) costs
- Fuel cycle costs, normalized per unit of energy produced
- Levelized unit cost of energy.

In the development of the MSBR cost estimates were determined along with the cost estimate for a corresponding PWR (in 1971). The cost estimate for the MSBR was re-evaluated as part of recent (FY-2010) fuel cycle options studies at ORNL and the life cycle costs escalated to a recent date (2010) with the details provided in the paragraphs below. This economics analysis has been performed to provide a levelized cost of electricity unit cost (LCOE) based on scaling of detailed cost estimates developed as part of the MSBR program and life cycle cost levelization methodology based on the G4-ECONS model.

The only other cost study found of recent vintage was a levelized electricity cost study by Ralph Moir of Livermore National Laboratory and published in *Nuclear Technology* (Moir 2002). Both studies are summarized in the Table R7-2 below

Table R7-2. Reference costs from two recent MSR Life Cycle Cost Analyses (2012 const $ assumed ~ 2010 const $).

	Low	Reference NOAK	High
(Moir 2002) 1000MWe MSR [Const yr 2000$]	N/A	Overnight capital; $1548/KWe LCOE: $38.4/MWh	N/A
(ORNL 2010 unpublished) 1000MWe {2010$]	N/A	Overnight: $4000/KWe LCOE: $63.5/MWh	N/A

Historical Estimates. Cost estimation of the MSR has the benefit of a reasonably comprehensive conceptual design and cost study done for a full-sized (1000 MWe) plant prepared shortly after the MSRE at Oak Ridge was shut down. This 180 page report, *Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor* was issued in June 1971 as ORNL-4541 [Robertson 1971]. This report contains process layouts, chemical flowsheets, energy and material balances, commodity lists, equipment descriptions, and life cycle costs for a complete fuel cycle system including both the reactor plant and a

self-contained chemical processing plant for treatment of the salt bearing the dissolved fission products and other fuel components. Availability of this ORNL report made it possible to prepare an updated parametric life cycle cost estimate based on the following economic and institutional changes from 1970 (when the estimate was prepared) to the present day (2010):

1. General escalation/inflation
2. A much more stringent regulatory environment
3. Greater quality assurance and inspection requirements
4. Permitting costs reflecting more stringent environmental regulations
5. Increased recognition of non-proliferation requirements.

Fortunately, at the time the MSR estimate was prepared, there was also available detailed capital cost information on the 1970 vintage PWR prepared by the same estimating team. A side-by-side, account-by-account comparison is printed in ORNL-4541[Robertson 1971] as Table 15.1. Since we know how to what extent the life cycle costs for today's new PWRs have increased since 1970, we can examine the factors behind the increase (beyond general inflation) which might also be applicable to a MSR today.

Table R7-3 shows the rolled up comparative costs prior to preparation of this 2010 MSR estimate. All of the costs, including the levelized "mills/kW-h" busbar electricity generation costs, in the "1970 MSR" column come directly from the ORNL-4541 report and are reported in 1970 constant dollars. The "1970 PWR" column includes PWR capital costs from the ORNL MSR report. The other PWR figures of merit given (capacity, factor, levelized electricity costs, etc.) are based on actual cost experience in the early 1970's for this vintage 1000-MWe PWR and are also in constant 1970 dollars. The third column, 2010 PWR cost, is based on projections for today's Generation III+ PWR of comparable size, such as the Westinghouse AP-1000. The values are in today's (2010) constant dollars. The levelization technique used to obtain unit electricity costs for the 2nd and 3rd columns is described in the *Global Nuclear Energy Partnership Economic Tools, Algorithms, and Methodologies Report* [Shropshire 2009a]. Economic parameters for the 2010 PWR, including its "once-through" fuel cycle, are presented in the 2009 *Advanced Fuel Cycle Cost Basis* report [Shropshire 2009b]. Along with the costs of labor and costs of equipment, commodity (concrete, steel, etc.) quantities became the basis of a "bottom-up" estimate for the reactor "overnight" cost, which included direct, indirect, owner's costs and contingency costs. The "all-in" cost includes the cost of financing (interest during construction) for the reactor project, and sums up all the costs incurred prior to commercial operation. Such "bottom-up" estimates were prepared for both the MSR and PWR in 1970.

The 2010 PWR costs are based on adjustments to financial data submitted by utilities to economic regulators. ([Shropshire 2009a] and Module R1 of this update reference the sources of much of this data). No detailed "bottom-up" estimates are publicly available for these plants because of the proprietary nature of such data. It should be noted that the costs in Table R7-10 are for "Nth-of-a-kind" or NOAK reactors for which the benefits of licensing, construction, and operational learning relative to the "first-of-a-kind" or FOAK units have been realized. Many of the LWR reactor projects now being licensed are essentially FOAK projects and all-in capital costs in the range of $3500 to $8000/kWe are predicted.

Table R7-3. Summary Nth-of-a-Kind (NOAK) Cost Data Available Prior to Preparation of 2010 Parametric Estimate.

Attribute	1970 MSR	1970 PWR	2010 PWR
Reactor net electric power capacity (Mwe)	**1000**	**1000**	1000
Capacity factor	**80%**	60%	90%
A II-in Capital Cost ($/kWe) (Const yr-of-est $)	**202.6**	**200.7**	4000
Plant economic life (for capital recovery)	**20**	20	60
Plant Regulatory life	**40**	40	60
Levelized costs			
Capital Recovery	**4.0**	3.7	38.6
O&M incl capital upgrades & replaceables	**0.5**	0.9	10.2
Fuel Cycle (MSR includes proc. eqt cap recovery)	**08.**	.9	5.9
D&D of reactor	**0.0**	0.1	1.0
Total	**5.3**	7.6	55.7
Carbon steel required[1] (MT) for construction	**2333.8**	40000	42000
Concrete required (cubic meters)	**79696**	75026	
Hastelloy-N required (lb) for construction	**638000**	0	0
Graphite required (lb) for construction	**730000**	0	0
1. MSR# does not include containment or structural steel (bolded values are from ORNL-4541)			

It should also be noted that the costs considered in the 1970 ORNL study and this study do not include the costs to develop and demonstrate MSR technology, nor the costs of prototype or first-of-a-kind (FOAK) plants.

Approach to Preparation of 2010 MSR Estimate. The cost estimate in ORNL-4541 was examined in four different areas: capital cost, non-fuel operations and maintenance (O&M) costs, fuel cycle costs, and decommissioning costs. Each of these is addressed below. It is first necessary, however, to identify and list some base assumptions regarding the hypothetical MSR which have effects throughout the estimate. These include the following:

1. Thermal Power: 2250 MWth; Net Electrical capacity: 1000 MWe;
2. Thermodynamic efficiency: 44%
3. Electricity production 7 billion kilowatt-hours per year
4. Capacity factor: 80%
5. Fuel cycle: Closed thorium/U-233 cycle with both fertile and fissile materials incorporated in a single fluid molten salt mixture
6. Breeding ratio : 1.06; Fissile yield: 3.3%; Doubling time 22 yrs
7. Steam Rankine Cycle with single turbine/generator
8. Salt composition: 71.7 mole% LiF (Li-7 enriched); 16.0 mole % BeF2;
9. 12.0 mole % ThF4; 0.3 mole % UF4 (mostly U-235)
10. Material balance basis: equilibrium cycle established, only fertile ThF4 feed
11. Moderator: Graphite; core components replaced every 4 years
12. Reactor cell diameter and height: 72 ft dia; 42 ft high

13. Average overall core power density: 22.2 kW/liter
14. Salt temperature: ~700C
15. Startup fissile material assumed in ORNL-4541 report: U-233 from a stockpile
16. Startup fissile material assumed for this report: 19.95% U-235 LEU (alternate assumption made for non-proliferation considerations)
17. Chemical processing cell located in same building as reactor; costs of chemical processing equipment, but not hot cell structure, included in fuel cycle cost.
18. Percent of salt replaced annually: 6.67% of initial inventory incl ~8000 kg ThF_4
19. Ownership and plant financial environment: Regulated private utility
20. Cost basis: Nth-of-a-kind plant
21. MSR fleet size assumed for sizing of support facilities such as lithium-7 enrichment and salt purify
22. Containment dome: Yes: 134 ft diameter, 189 ft height.

Plant Capital Cost. Since the authors of ORNL-4541 found that the capital cost per kilowatt for the MSR was nearly identical to that of the PWR, it is useful to see if that finding would still hold today. The first step taken in this analysis was to escalate the PWR cost of ~200$/kWe in ORNL-4541 to today's dollars. Using the Handy-Whitman [Whitman 2008] nuclear construction cost index from 1970 to 2009 a factor of 6.7 results, giving a value of $1350/kWe for today's PWR. This value is known to be low by a factor of ~3 for the projected NOAK 1000-MWe class PWR's being considered for deployment today. (This 2012 update to the *Advanced Fuel Cycle Cost Basis* report suggests a range of $2300/kWe to $5800/kWe for the NOAK overnight cost of a generic LWR. This implies that non-direct construction factors other than inflation have driven up capital costs since 1970. Among these factors would be regulatory, financial, schedule, start-up, quality assurance, ES&H, and security related requirements and factors which have changed from 1970 or did not even exist in 1970. For the following reason it is assumed that these factors would affect the MSR the same way the PWR was affected. The following arguments are presented in this regard:

- Despite the low pressure of the MSR system, the containment must be designed to withstand external threats such as an airplane crash. This is a requirement which has changed since 1970.
- In addition to being a nuclear plant, the MSR is also a small radiochemical plant operating in a hot cell. In addition to nuclear-related hazards such as criticality and radiotoxicity, there are also concerns associated with high-temperature operations, such as fire safety, equipment reliability, and treatment, packaging, and disposition of high level waste in halide or halide-derived chemical forms.
- The construction of the MSR requires the use of additional materials for which the unit costs (cost per unit mass) are likely to be higher than for the typical materials used in LWRs and their cores, such as carbon steel, stainless steel, and zircalloy. Among the structural materials are high purity, high temperature graphites and Hastelloy-N. (The special salts will be discussed under fuel cycle costs.) Initial analysis indicates that the unit costs for these materials, especially the graphite, have risen considerably from the values used in the 1970 ORNL study. Table R7-4 shows the unit costs and material requirements for the MSR and PWR.
- The 1970 estimate had MSR indirect costs as only 35% of direct costs. For most of today's nuclear projects, these "people-related" indirect costs are now a much higher fraction of or are equal to the direct construction costs. This is a result of today's regulator-mandated requirements for quality control, design certification, inspections, etc.

Table R7-4. Construction commodities and unit costs.[1]

Material	Quantity for 1000 Mwe MSR Construction and Initial Inventory	Mass Units	1970 Unit Cost	2010 Unit Cost
Li-7 Enriched Lithium Fluoride Salt	60816	kg LiF	33.1	473.0
Beryllium difluoride salt	24430	kg BeF2	16.5	135.7
Thorium as Thorium Tetrafluoride	90540	kg Th	8.6	65.0
Hastelloy-N	638000	lb	18.2	20.0
Graphite	730000	lb	10.7	80.0
Carbon Steel (non-strucutral, non-containmnet)	2334	MT	1323	500.0
Concrete	49696	cubic meters	134.7	98.0
1. Bolded values are from ORNL-4541.				

There are also factors which can have a beneficial effect on capital costs vis-a-vis the LWR:

- Because of the high (~700C) coolant temperature, the MSR could be adapted to a more efficient Brayton cycle utilizing a gaseous working fluid. The higher thermodynamic efficiency would lower the capital component of the overall electricity cost by allowing more electricity production from the same 2250 MWth thermal power capacity.
- Lower plant operating pressure results in equipment metal fabrication thicknesses and weights that are lower than those for PWR components. This would allow more fabrication to be done domestically and in regional foundries. Transportation cost to the plant site would also be greatly reduced.
- The MSR concept is readily adaptable to the "small reactor" more flexible siting requirements now envisioned for LWRs such as the NuScale© and mPower© concepts.

It is arguable that are no reasons or "show-stoppers" that would cause the overall specific capital cost ($/kWe) of an NOAK MSR to significantly exceed that for the PWR. The ranges for the NOAK LWR suggested in the 2009 and 2012 updated Advanced Fuel Cycle Cost Basis report should be applicable to the NOAK MSR. These are:

	2009	2012
Low	2300 $/kWe	2300
Nominal (most likely)	3500 $/kWe	N/A
High	5000 $/kWe	5800
Suggested MSR value	4000 $/kWe	N/A

This means that the recommended "all-in" cost for a 1000-MWe NOAK MSR would be around $4 billion. It should be noted that for the current analysis the $/KWe cost for all of the "raw" or "base" construction commodities, not including the salts, would be less than $100/KWe. This is also true of PWRs, for which an analysis by Per Peterson of UC-Berkeley [Peterson 2009] shows are on the order of $35/kWe. This means that most of overall direct material costs for construction of any kind of reactor are "value-added" costs in going from raw materials to specifically fabricated parts or equipment, or materials requiring special installation requirements such as "nuclear-grade" concrete. These "people-related" or costs with high labor content will be present regardless of the selected reactor technology.

MSR NON-FUEL O&M COSTS. ORNL 4541 presents a list of calculated annual O&M costs and a list of major MSR materials requiring periodic replacement. The ORNL-4541 Tables are reproduced below as Table R7-5 below.

Table R7-5. Annual costs from Cost estimate in ORNL 4541 (Robertson 1971).

Table D.15. Cost of replacing reactor core assemblies in the MSBR

In thousands of dollars

Cost of assembly	
Hastelloy N – see Table D.4	1,092
Graphite – see Table D.5	3,753
	4,845
Chargeable power revenue loss during core assembly replacement[a]	
Special labor cost per replacement[b]	500
Total cost per replacement	5,345
Effect on power production cost, mills/kWhr[c]	0.17

[a]It is assumed that the MSBR core assembly can be replaced during the plant downtimes for inspection and repair of other equipment, such as the turbine-generator, which are accommodated by the 80% plant factor, and no additional plant outage is chargeable against core replacement.

[b]The labor force for making core replacements is assumed to be in addition to the normal plant operating and maintenance crew.

[c]While various methods could be used to estimate the cost of future core replacements, a sufficiently representative and straightforward method is to assume an extra amount charged per kilowatt-hour, which is set aside, at 8% interest compounded annually, so that at the end of four years the total cost of a replacement will have been accumulated.

Table D.16. Estimated annual costs for plant operation and maintenance[a]

Staff payroll[b]	$ 800,000
Fringe benefits[b]	80,000
Subtotal – plant staffing	880,000
Consumable supplies and equipment	400,000
Outside support services	140,000
Miscellaneous	80,000
Subtotal	1,500,000
General and administrative	225,000
Coolant-salt makeup[c]	9,000
Nuclear liability insurance	
Commercial coverage (net)	240,000
Federal Government coverage	67,500
Total direct annual cost	2,061,500
Fixed charges on operation and maintenance working capital	38,800
Total annual cost	$2,080,300
Contribution to power cost[d]	0.30 mill/kWhr

[a]Based on cost breakdown and computation prescribed in NUS-531 (ref. 119). The values agree reasonably well with those reported by Susskind and Raseman (ref. 121). Costs do not include chemical processing, which is included in the fuel-cycle cost, nor special costs associated with periodic replacement of the core graphite.

[b]Based on NUS-531 (ref. 119) recommended values for July 1968 escalated 8%.

[c]Makeup cost assumed to be 2% of inventory.

It can be seen that the total O&M contribution to the busbar cost totals 0.47 mills/kW-h in 1970 constant dollars. The comparable number for today's PWR would be over 20 times higher at ~10 mills/kW-h. There are several factors which have caused this to increase at a rate greater than general inflation. For example the 1970 estimate above has $888K/year for staffing costs (not including the chemical plant). This average loaded salary of $11,000/per person per yr (1970) represents a staff of 80. ES&H, security, regulatory, and training-related requirements have driven the number of people required to operate a 1000-MWe nuclear plant of any type to a few hundred permanent staff and more could reasonably be expected because fuel processing plant staff is required. For the 2010 MSR estimate a staff of 250 is assumed at an average loaded salary of $110,000/ per person per year. Table R7-6 shows the other adjusted O&M costs. These are taken from an estimate [Gen IV 2007] for today's PWR in the 1000-MWe size class.

Table R7-6. Annual O&M and Non-Salt Material Replacement Costs (Const 2010 dollars).

O&M Categories					Annual $M	mills/kwh
Staff Payroll incl fringes (250 people @ 110K @ ave)					27.500	
Consumables					21.500	
Subcontracts & miscellaneous					5.000	
G&A incl regulation					11.000	
Salt make-up (2% of inventory)					0.642	
Insurance (private and federal) + taxes					6.000	
Charges on working capital					0.000	
Subtotal					71.642	10.22
Major core structures replaced every 4 years						
Labor cost			5.00	div by 4	1.25	0.18
Graphite core structures	351382 lb	80 $/lb	28.11	div by 4	7.03	1.00
Hastelloy-N structures	134830 lb	20 $/lb	2.70	div by 4	0.67	0.10
					8.95	1.28
TOTAL ANNUAL O&M incl REPLACEABLES						**10.31**

The costs for replaceables utilize the unit commodity costs assumed in Table R 7.4 for Hastelloy-N and graphite. A major uncertainty in the annual costs is the handling of waste and the staffing involved. Waste management was not substantially addressed in ORNL-4541.

MSR Fuel Cycle Costs. The major claimed cost advantage for the MSR over other reactor types is in fuel cycle costs. In the 1970 estimate, the advantage from the fuel cycle component of the busbar electricity cost was over 4 to 1 (0.8 mills/kW-h for the MSR and 2.9 mills/kW-h for the PWR [the latter number based on actual PWR experience]). From the 2010 analysis there still appears to be an advantage on the order of 30% in terms of busbar cost for the MSR against the PWR operating on a UOX once-through cycle. The escalation above inflation observed for O&M and capital costs is assumed to not exist for the fuel cycle steps. The cost advantage exists for the following reasons:

- Since the MSR is assumed to be an "equilibrium" breeder, only relatively inexpensive ($65/kgTh) fertile material is required. PWRs are operated on low-enriched UOX costing at least $1600/kgU for fresh reload UOX fuel assemblies. The source material utilization for PWRs is less than 1%, i.e. over 99% of the initial mined U ends up either in the spent fuel or in enrichment plant tails. The source material utilization for mined thorium is essentially 100% or higher in the breeder mode.
- The only fissile required is the 19.95% U-235 material required for reactor startup. The cost of this material (over $13,000/kgU) is amortized over many years of operation.
- The molten salt reactor concepts eliminates the need for separate fuel fabrication and reprocessing services, and transportation to and from the reactor for fresh and spent fuel assemblies. It incorporates a totally "integral" fuel cycle.
- The fissile inventory of the overall systems, and the annual fertile makeup requirement, is very low. Just over 8000 kg of ThF4 per year are needed to sustain the reactor.

The fuel cycle assumptions and cost implications used in this analysis are as follows:

1. The material balances for both the initial inventory and salt replacement are complex and the extensive material balance tables and flowsheets from ORNL-4541 are not repeated here.
2. The LiF salt must be over 99.9% enriched in the isotope lithium-7. No large scale facility exists for this purpose, nor is a design and cost estimate for such a facility available as of 2010. The "natural" lithium product should be readily available at $4000 to $6000 per ton of lithium carbonate (Li_2CO_3). Discussions with isotope experts at ORNL indicate that the 99.99% Li-7 material could cost as much as $6 per gram Li, but this would be for kilogram type quantities. If a large plant (700 MT Li/yr of feed and 190 Kg of enriched Li product) capable of producing this material for the initial inventories and makeup salt for 32 GWe of MSR capacity could be constructed, the unit cost ($/g of enriched Li) should scale downward.
3. A beryllium salt, BeF2, is also required. Its unit cost has increased significantly from the 1970 estimate. This is probably due to its use as a strategic defense material and the significant ES&H issues associated with its handling.
4. The overall "flibe" salt must be very pure. A purification step is required, but no costs for this were presented in the 1970 estimate. An arbitrary cost of $100/kg of salt is assumed for a large facility servicing multiple MSRs.
5. The long term forward average price for thorium of $65/kgTh is from the 2009 *Advanced Fuel Cycle Cost Basis* report [Shropshire 2009a]. (See Module A2 for updated values and ranges.)
6. The 1970 report did not include waste management costs such as conditioning, packaging, and disposal costs for GTCC and high level fission product wastes. For this 2010 analysis unit waste handling costs from the 2009 Advanced Fuel Cycle Cost Basis report were applied to particular classes of fission products, such as volatiles, alkali metals, etc.

Table R7-7 itemizes the unit material costs used in both the 1970 and 2010 studies. Costs are normalized to $/kg where possible.

Table R7-7. Unit Cost Values for 1970 and 2010 MSR Cost Studies.

Item	1970$ Cost Value	Measurement Units	1970 Cost in $/kg	2010$ Cost Value	Measurement Units	2010 Cost in $/kg
Enriched LiF salt	15	$/lb LiF	33.1	473	$/kgLi	
Beryllium salt	7.5	$/lb BeF2	16.5	136	$/kgBeF2	
Thorium salt	6.5	$/lb ThF4	14.3	65.0	$/kgTh	
U-233	13	$/gram U as UF4	13000	n/a		
Pa-233	13	$/gram Pa as PaF4	13000	n/a		
93.5% U-235	11.2	$/gram U as UF4	11200	n/a		
19.95% U-235				13781	$/kgU	
Graphite	11	$/lb	24.255	80	$/lb	176.40
Hastelloy-N	8 to 38	$/lb (14 used)	18 to 80	20	$/lb	44.10
Pu-239	9.3	$/g Pu	9300	n/a		
Salt purification			-	100	$/kg salt	
Carbon steel	0.6	$/lb	1.32	500	$/MT	0.50
Concrete	103	$ per yd3 (installed)		98	$/m3	n/a
Stainless steel	1.2	$/lb	2.65	3000	$/ton	3.31

Annual recurring costs for the fuel cycle materials are calculated by multiplying the unit cost ($/kg) times the annual “make-up” requirement (kg/yr) from the material balance. The costs for initial inventories are annualized and recovered using a fixed charge rate of 13.7%. The $13.5M 1970 cost for chemical process equipment has been escalated to $135M in 2010 dollars and is distributed over all of the operating years by use of a 13.7% fixed charge rate. Table R7.8 shows a summary of the fuel cycle costs in both millions of dollars per year and as a component of the overall busbar cost of electricity in mills/kW-h. The 2010 calculated value of 4.2 mills/kW-h is lower than the unit cost projected for PWRs operating on either a once through or partial recycle mode.

Table R7-8. Summary of Fuel Cycle Costs for 1970 and 2010 Estimates.

RE-CREATION of 1970 ESTIMATE		Amount		Unit cost	Cost Units	Total cost ($M)	CRF	Annual cost ($M/yr)	Unit cost (mills/kwh)
Capital Recovery of Initial Inventories (A+B from Table D.2 of ORNL 4541)									
Enriched Li salt (99.99% Li-7) as LiF		16373	kg Li	123	$/kg Li	2.01	0.1320	0.27	0.04
Beryllium salt as BeF2		24430	kg BeF2	16.54	$/kgBeF2	0.40	0.1320	0.05	0.01
Salt purification		85246	kg (LiF+BeF2)	0.00	$/kg salt	0.00	0.1320	0.00	0.00
Initial Thorium load (as ThF4)		90540	kg Th	19.03	$/kg Th	1.72	0.1320	0.23	0.03
Initial Uranium-233 (as UF4)		1286	kg U-233	13000	$/kgU	16.72	0.1320	2.21	0.31
Initial Protactinium-233 (as PaF4)		110	kg Pa	13000	$/kg Pa	1.43	0.1320	0.19	0.03
Initial Uranium-235 (as VHEUF4)		112	Kg U-235	11200	$/kgU	1.25	0.1320	0.17	0.02
Subtotal						23.54		3.11	**0.44**
Capital Recovery of Chem proc Eqt incl indirects (Item D from Table D.2)				13.5	$M		0.137	1.85	**0.26**
Annual Fuel Cycle Material & Service Usage (Item C from Table D2)									
Enriched Li salt (99.99% Li-7) as LiF		1092	kg Li/y	123	$/kgLI	0.134	---	0.13	0.02
Beryllium salt as BeF2		1629	kg Be/y	16.54	$/kgBeF2	0.027	---	0.03	0.00
Salt purification		5683	kg salt	0.00	$/kg salt	0.000	---	0.00	0.00
Thorium load (as ThF4)		8005	kg ThF4/y	14.33	$/kgThF4	0.115	---	0.11	0.02
Subtotal								0.28	**0.04**
O&M Costs for Chemical processing System (Item E from Table D2)						0.70	---	0.70	**0.10**
Production Credit									**-0.09**
TOTAL FUEL CYCLE COST						---			**0.76**
RE-Estimate with 2010 Unit Costs and no HEU									
FUEL CYCLE: Capital Recovery of Initial Inventories		Amount		Unit cost	Cost Units	Total cost ($M)	CRF	Annual cost ($M/yr)	Unit cost (mills/kwh)
Enriched Li salt (99.99% Li-7) as LiF		16373	kg Li	1757	$/kg Li	28.77	0.0847	2.44	0.35
Beryllium salt as BeF2		24430	kg BeF2	136	$/kgBeF2	3.32	0.0847	0.28	0.04
Salt purification		85246	kg (LiF+BeF2)	100	$/kg salt	8.52	0.0847	0.72	0.10
Initial Thorium load (as ThF4)		90540	kg Th	65	$/kg Th	5.89	0.0847	0.50	0.07
Initial Enriched Uranium (as UF4)		7522	kg U	13781	$/kgU	103.67	0.0847	8.78	1.25
(assume 19.95% U-235); req'd fissile (235) for Start Up is:	1500.72								
Subtotal								**12.71**	**1.81**
Capital Recovery of Chem proc Eqt (unrec cost 10x 1970 #) (no production credit)				135	$M		0.0847	**11.43**	**1.63**
Annual Fuel Cycle Material & ServiceUsage									
% of loaded salt inventory replaced per year =	6.67%								
Enriched Li salt (99.99% Li-7) as LiF		1092	kg Li/y	1757	$/kgLI	1.92	---	1.92	0.27
Beryllium salt as BeF2		1629	kg BeF2/y	136	$/kgBeF2	0.22	---	0.22	0.03
Salt purification		5683	kg salt	100	$/kg salt	0.57	---	0.57	0.08
Thorium load (as ThF4)		6030	kg Th/y	65.00	$/kgTh	0.39	---	0.39	0.06
Subtotal								**3.10**	**0.44**
Waste management incl geologic disposition (what is not incl in O&M)									
Volatile FPs		54	kg FP/yr	22500	$/kg FP			1.22	0.17
Alkali-metal FPs		51	kg FP/yr	6500	$/kgFP			0.33	0.05
Noble & Lanthanide FPs		120	kg FP/yr	6500	$/kgFP			0.78	0.11
								2.33	**0.33**
TOTAL FUEL CYCLE COST									**4.22**

Decontamination and Decommissioning Costs (D&D). End of life reactor D&D costs are usually calculated as a fraction of the direct costs if no detailed D&D estimate is available. The 1970 estimate did not include any D&D costs. D&D for this study is assumed to cost 25% of the assumed "all-in" capital cost or $1 billion in constant 2010 dollars. An escrow or sinking fund is collected annually during operations such that this amount is available at end of life to cover these costs. If a 7.5% discount rate is assumed, a little over $4M per year is required. The busbar cost component for this category is 0.6 mills/kW-h.

Total Life Cycle Busbar Cost or Levelized Cost of Electricity (LCOE). Table R7-9 Summarizes the "Mills/KW-h" or "$/MW-h" unit electricity generation cost results from the 1970 estimate and this study. The same figures of merit for today's new 1000-Mwe class PWR are also shown for comparison. The PWR is assumed to have a 60 year life and amortization period. The MSR is assumed to have a 40 year life and amortization period (although future MSRs would certainly be designed for 60 year lifetimes, the original MSRs designs considered a shorter lifetime). This account for the lower capital component for the PWR; however both are assessed at $4000/kWe.

Given the many uncertainties in the estimates, the 2010 MSR estimate falls well within the range projected for new PWRs. At this time no economic "show-stoppers" have been identified that would make this concept non-competitive. This study did not incorporate a complete uncertainty analysis, thus "deterministic" or single point values were used. In reality uncertainty ranges exist for all parameters and additional effort in the area of uncertainty analysis is recommended.

Table R7-9. Comparison of Busbar Generation Costs.

	1970 MSR	2010 MSR	2010 PWR
Capital	4.0	48.3	38.5
O&M	0.5	10.3	10.2
Fuel Cycle	0.8	4.2	5.9
D&D	0.0	0.6	2.0
Total (mills/kW-h)	5.3	63.5	56.6

R7-7. DATA LIMITATIONS

Molten salt reactors were considered to be attractive power producers because of favorable economic, fuel utilization, and safety characteristics. Rosenthal et al. were optimistic in their 1970 assessment [Rosenthal 1970]:

> *"The avoidance of fuel fabrication, the ease of processing, and the low fissile inventory should result in low fuel cycle costs."*

It was thought that the cost of handling radioactive fluids in conjunction with reactor operation would be offset by other compelling features, including, higher temperature leading to increased thermal efficiency, low pressure operation, significant safety margin related to boiling margin. However, limited economic analyses of the MSR concepts have been performed.

The following issues represent significant cost uncertainties for extrapolating MSRE experience to a power reactor:

1. The cost of handling radioactive fluids at the reactor site. Many institutional issues also exist,
2. The flexibility to startup with ^{233}U, ^{235}U or Pu allows the core to be started with the most economic (or prudent) of the available options.
3. The availability of high temperature power conversion systems to utilize the available temperature of salt fueled reactors.
4. Length of time graphite can be used in the vessel (neutron damage) or in contact with the fuel salt (contamination) before requiring replacement.
5. Cost of salts and the ability process them for continuous use compared to the need to replace them periodically.
6. Capital costs, including the reactor and salt processing equipment.

7. Material costs, particularly the increased costs of nickel-based alloys compared to that of stainless steel.
8. Radiation induced embrittlement of Hastelloy-N was a concern for extrapolating that material from use on the MSRE to the MSBR and a modified version of the alloy using titanium or hafnium as a stabilizer was thought to overcome this issue.
9. Graphite dimensional changes with flux and temperature were a significant concern for a power reactor application requiring periodic replacement of the graphite moderator

R7-8. COST SUMMARIES

A summary of the cost parameters developed from the recent ORNL study are provided in Table R7-10. These results indicate that the MSBR has a lower fuel cycle cost and a larger levelized electricity cost (~10%). The primary driver for the larger levelized electricity costs is the assumption that the PWR lifetime is 60 years (as is the current experience), while the MSBR is assumed to be 40 years. With the same reactor lifetime the overall levelized electricity cost is essentially the same for the MSBR and PWR. No cost estimates were prepared for the DMSR, although it would be expected to have a lower capital cost than the MSBR because of the lack of a need for the fuel salt processing system. The DMSR would be expected to have a larger fuel cycle cost because of the need for enriched uranium.

Table R7-10. Comparison of Fuel Cycle Component and Total LCOE of MSR and PWR systems.

	MSR (1970)	MSR (2010)	PWR (2010)
Fuel Cycle Costs (mills/kWe-hr)	0.8	4.2	5.9
Levelized Electricity Cost (mills/kWe-h)	5.3	63.5	56.6

- An update of economics evaluation performed for the MSBR indicates a levelized electricity cost that is approximately 10% higher than the LWR reference. With a comparable reactor lifetime, the MSBR would have a levelized electricity cost that is comparable to the LWR reference. The DMSR.
- The MSR has safety advantages and disadvantages in comparison with the LWR reference. The advantages include low excess reactivity, lower radionuclide inventory, low pressure system, passive decay heat removal, fuel is already molten – fuel meltdown accident is not possible. Disadvantages include contamination of primary system, the enhanced production of tritium, high temperature operation, and chemical hazards associated with fuel processing system.
- While two molten salt reactors have been built and operated, the average technology readiness level can be characterized as “proof of principle” (TRL 5-6) and considerable research and development as well as significant amount of technology development will be required to bring the system to the level of commercial operation.

For comparison to other reactors in the “R” Modules a “What-it-Takes” overnight cost range is required. The values selected in Table R7-11 are based partly on the recent ORNL estimate above.

Table R7-11. “What-It-Takes” Specific Overnight Cost for Liquid-Fueled Molten-Salt Reactor.

What-It-Takes Overnight Cost:	Low (2012$)	Nominal (2012$)	High (2012$)
Thermal MSR System	2200	5500	9000

Assigning a range to this value is difficult because no uncertainty analysis was performed as part of the recent ORNL study. The low value selected here was based on taking the Moir study (Moir 2002) specific overnight cost of $1584/kwe and escalating it to 2012$ using the escalation Table at the beginning of the 2012 Update report. A value of $2200/kWe results. (Moir’s study seems to be based on optimistic parameters, such as a capacity factor of 90%.). Since this reactor concept is so different from solid-fueled concepts, and has such a large-component of its cost dedicated to chemical systems, the

capital cost risk is very high. This is especially true given the recent cost experience with nuclear chemical facilities such as reprocessing plants. For this reason a high overnight cost value greater than the other R-Module reactor types was assigned. The nominal value is assumed to lie approximately in the middle of this range.

For 2015, the values in Table R7-11 were escalated from 2012$ to 2017$ using an escalation factor of 1.088 from the table at the beginning of this report followed by appropriate rounding. The following" What it Takes" Table R7-12 results:

Table R7-12. "What-It-Takes" Specific Overnight Cost for Liquid-Fueled Molten-Salt Reactor.

What-It-Takes Overnight Cost:	Low (2015$)	High (2015$)	Mean (2015$)	Mode (2015$)
Thermal MSR System	2400	9800	6100	6000

Figure R7-4 below shows the uncertainty distribution and parameter for the thermal MSR evaluated in this module:

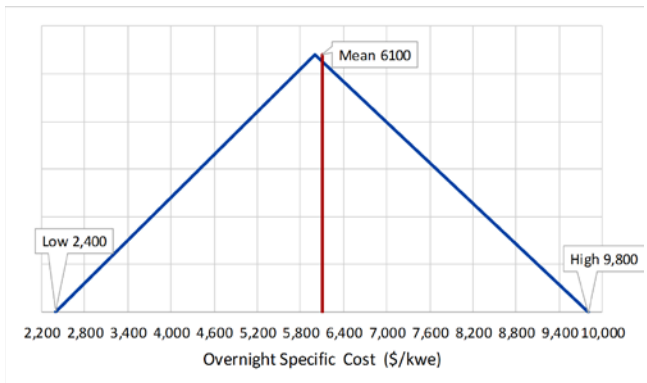

Figure R7-4. Distribution and parameters for MSR specific overnight cost

The work performed here represents a very preliminary review and assessment of MSR technology. Proposed future work include:

- Assessment of a minor actinide burning MSR, such has been proposed elsewhere, as an additional modified open cycle concept. With these systems, the reactor is fueled with used LWR fuel with the fuel circulated to achieve a high burnup of actinides. Both thermal spectrum and fast spectrum systems should be considered along with the used LWR fuel processing based on fluoride volatility.

A report on the design aspects of the Fast Spectrum MSR options was recently prepared at ORNL (Holcomb 2011). No cost estimate was prepared as part of this study; however, some quantitative economic conclusions might be drawn by comparing recent reports.

R7-9. SENSITIVITY AND UNCERTAINTY ANALYSES

No sensitivity studies have been recently undertaken for MSR systems. This will likely become an objective for further MSR System Assessment tasks.

R7-10. REFERENCES

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Module R8

Solid-Fueled Salt-Cooled Reactors

Module R8

Solid-Fueled Salt-Cooled Reactors

R8.MD. SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2012 AFC-CBR)
- **Estimating Methodology for latest (2012 AFC-CBR) technical update from which this 2017 update was escalated:** Examples salt-cooled HTR estimates from ORNL were utilized to develop cost estimates combined with engineering judgment of the differences with LWRs. Some bottom-up estimating data was available.

R8.RH. REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2012 as Module R8.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2012
- **New technical/cost data which has recently become available and will benefit next revision:**
  - Not aware of any new cost information (Holcomb, et al. 2011).

R8-1. BASIC INFORMATION

Liquid-fueled Salt-cooled reactors, more **commonly referred to in the recent literature as Advanced High Temperature Reactors** (AHTRs) represent a class of reactors that involve the use of fluoride salts which serve as the reactor coolant material and can accommodate reactor coolant outlet temperatures of up to 750C. (It should be noted that a liquid-fueled, high temperature reactor has a molten-salt coolant that also contains the fuel. This reactor is discussed in Module R7 and is usually what is meant when the term "MSR" is used. The term AHTR might also be applied to advanced high-temperature gas-cooled reactors, which are discussed in Module R3. **For these reasons a better acronym for the subject of this module would be FHR (Fluoride-salt High-temperature Reactor).**

This reactor type borrows design characteristics from several reactor types:

- Coolant: Molten salt nearly identical to that for MSRs
- Fuel: High-temperature tolerant TRISO particle fuel used for HTGRs
- Moderator: Graphite moderator and fuel support structures (like prismatic HTGR)
- Reactor Vessel: Pool configuration similar to SFR
- Power cycle: Rankine (Brayton cycle is possible)

The reactor can produce both process heat (hot salt pumped to process) and electricity (using a high-efficiency Rankine steam cycle in the near term). Only one detailed cost estimate has been prepared for this reactor (Holcomb, Peretz, and Qualls 2011) and forms the basis for most of the information (reference case) in this module. Work on the FHR concept is ongoing at ORNL, the University of California at Berkeley, and other universities. The reference cost-estimate was developed using EEDB (Energy Economic Data Base) and G4-ECONS (Gen IV Excel Computation of Nuclear Systems) tools

developed at ORNL. Use of these tools and scaling relationships allows one to build up an estimate for an advanced reactor using detailed historical LWR cost information.

R8-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

Basic Reactor Features. The reference reactor operates at a high temperature, with the fuel salt core exit temperature in the range of 700 to 750 °C and includes a Rankine steam power conversion system with a 45% power conversion efficiency. The assumed thermal power capacity is ~3400 MW(th) and the electrical capacity ~1530 MWe. The system is passively safe and utilizes heat rejection via a Direct Reactor Auxiliary Cooling System (DRACS). The capacity factor is assumed to be similar to that of LWRs, i.e. ~90%. The fuel consists of enriched uranium (ceramic U compounds such as UOC or UO2) TRISO particles imbedded in graphite fuel plates. Until a more technically mature and economic method for reprocessing particle fuel emerges, this reactor would likely operate on a once-through fuel cycle.

R8-3. PICTURES, DIAGRAMS, AND DEPLOYMENT STATUS

Figure R8-1 below (from a University of California website) shows the basic reactor concept and how it can be utilized for electricity and/or process heat applications. Note that one application that garnered considerable interest was the use of process heat for hydrogen production. The reactor in the figure below is a pebble-bed example. Like the HTGR both pebble-bed and prismatic fuel configurations are possible. ORNL has concentrated on the prismatic option and UC Berkeley on the pebble-bed option. Figure R8-2 shows the prismatic plate fuel concept used for the reference reactor in the ORNL study. The fuel configuration is shown in Figures R8-3 and R8-4 below.

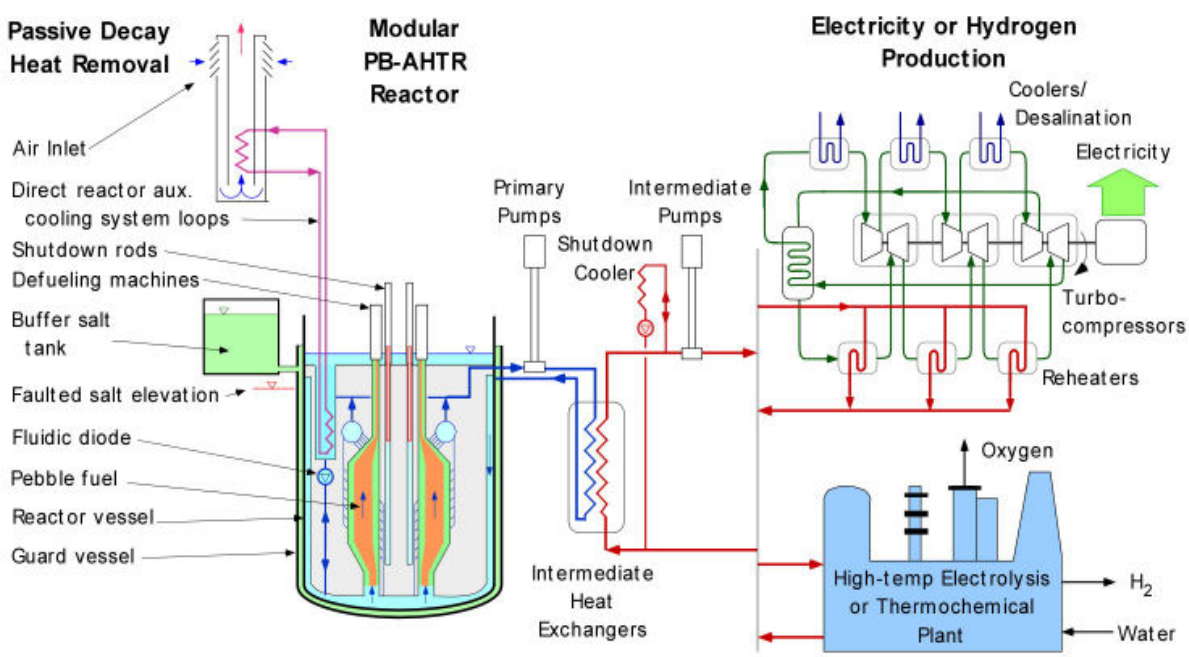

Figure R8-1. Basic ATHR-FHR Concept and its Applications (Pebble-bed example).

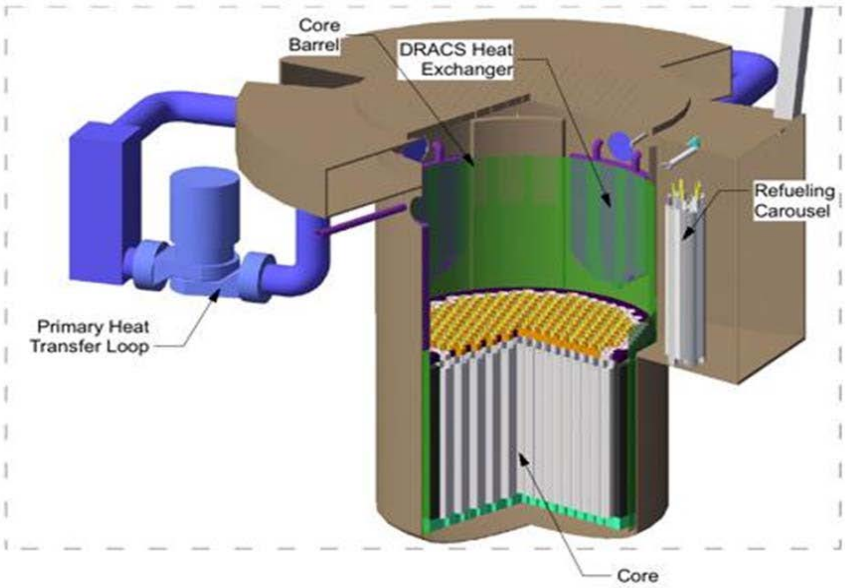

Figure R8-2. Prismatic-fueled reference AHTR-FHR concept.

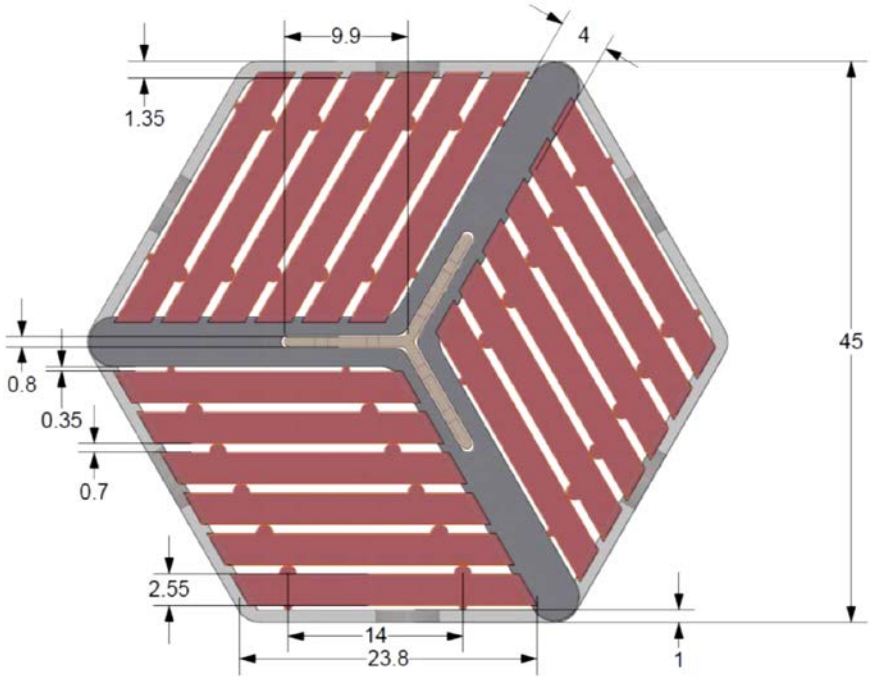

Figure R8-3. Top-down View of the reference AHTR fuel assembly showing beveled rectangular prism graphite fuel plates.

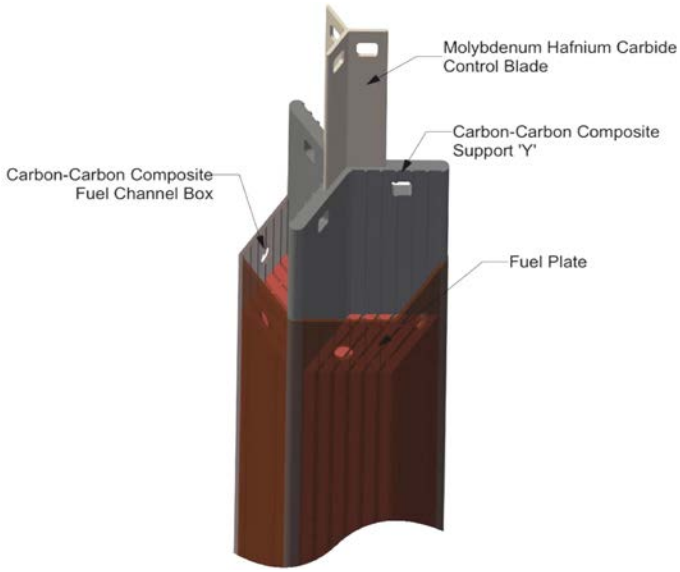

Figure R8-4. 3-dimensional side view of reference AHTR fuel assembly.

A major economics-influencing design factor for this reactor is the fact that a liquid-coolant allows much more heat to be removed from the TRISO fuel than if the coolant were a gas such as helium or carbon dioxide. A higher power density for the AHTR as compared to the HTGR is the result.

R8-4. MODULE INTERFACES

The interfaces would be much the same as for HTGRs and TRISO fuel (Modules R3 and D1-3). Treatment, packaging, and disposal of salt waste would require development and could benefit from work on Electrochemical Reprocessing (Module D2/F2).

R8-5. SCALING CONSIDERATIONS

Detailed information on scaling is described in the ORNL reference (Holcomb, Peretz, & Qualls 2011).

R8-6. COST BASES, ASSUMPTIONS, AND DATA SOURCES

The only data source to date on this reactor concept is the recent ORNL Technical Memorandum 2011/364 (Holcomb, Peretz, & Qualls 2011; http://info.ornl.gov/sites/publications/files/Pub32466.pdf). The following data in Table R8-1 are extracted from that report. Note that the authors started with a reference PWR to develop the estimate as was the case with the MSR in Module R7. G4-ECONS was used to calculate the LCOE, and the assumptions and details are in the reference report. Note that the specific cost is lower than for the reference LWR. The major contribution to this decrease is the higher thermodynamic efficiency of the FHR system (43%) as compared to the LWR (33%). This means that a reactor of similar size in terms of concrete and steel can produce significantly more electricity. Note that the specific costs shown in Table R8-1 below are based on the overnight cost. The ones presented in the ORNL report (Holcomb, et al.) are based on the “all-in” cost, which includes interest during construction.

Table R8-1. Comparison of Overnight and Total LCOE of FHR and PWR systems.

	FHR with 19.75% U-235 fuel (2010$)	FHR with 9.0% U-235 fuel (2010$)	System 80+ PWR (2001$)	1134 MWe PWR (2010$)
Overnight Cost incl initial fuel load ($/kWe)	2900	2700	<2000	3532
Levelized Electricity Cost (mills/kWe-h)	51.6	43.1	30.6	48.2

R8-7. LIMITATIONS OF COST DATA

There are many technical issues related to the salt and high-temperature materials which are still outstanding. These and other technical issues which influence cost are discussed in the MSR Module R7. The cost of the isotopically-enhanced lithium fluoride salt could be a major issue.

R8-8. COST SUMMARIES

Low, nominal, and high values are reported for the specific overnight cost, which suggests the use of a triangular distribution for uncertainty analysis. Table R8-2 shows a range for NOAK reactors which should be consistent with the other Reactor "R" Modules.

Assigning a range to this value is difficult because only a limited uncertainty analysis (single-variable sensitivities) was performed as part of the recent ORNL study. The low value selected here was based on taking the MSR low value (2200 $/kwe) downward to account for the more conventional reactor core and solid-fuel handling systems. The high value for the MSR ($9000/kwe) was also lowered to $8000/kwe, which is the high value for the HTGR (Module R3) for the same reason. The nominal value was chosen as the midpoint of this range.

Table R8-2. "What-It-Takes" Specific Overnight Cost for Solid-Fueled Molten-Salt Reactor.

What-It-Takes NOAK Overnight Cost:	Low (2012$)	Nominal (2012$)	High (2012$)
Molten Salt AHTR System	2000	5000	8000

For this 2017 AFC-CBD the year 2012$ amounts above were escalated to 2017$ using an escalation factor of 1.088. The resulting values were then rounded to maintain a proper number of significant digits. Table R8-3 results:

Table R8-2. "What-It-Takes" Specific Overnight Cost for Solid-Fueled Molten-Salt Reactor.

What-It-Takes NOAK Overnight Cost:	Low (2015$)	High (2015$)	Mean (2015$)	Mode (2015$)
Molten Salt AHTR System	2200	8700	5600	6000

Figure R8-5 below shows the probability distributions and parameters for this reactor type.

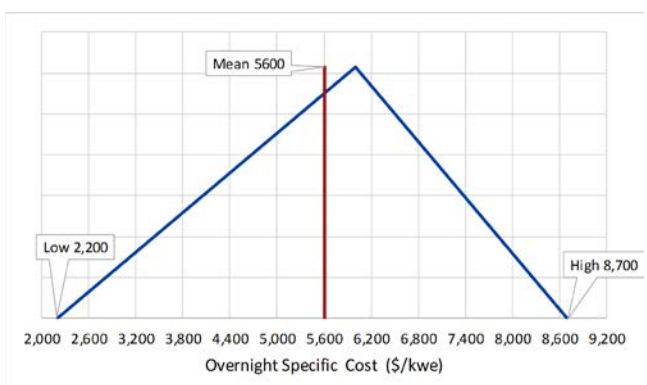

Figure R8-5. Distributions and parameters for Specific Capital Cost of Solid Fueled AHTR.

R8-9. SENSITIVITY AND UNCERTAINTY ANALYSES

A few sensitivity studies are described in the ORNL reference (Holcomb, Peretz, & Qualls 2011).

R8-10. REFERENCES

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R8-11. BIBLIOGRAPHY

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R9 Modules

Fission/Fusion Hybrid Systems

Module RP9

Preface to
Fission/Fusion Hybrid Systems

Module RP9

Preface to Fission/Fusion Hybrid Systems

RP9-1. BACKGROUND

Note: This module was added in 2012 as part of the AFC-CBD update. No new data has been added in the 2013-2015 time frame. Unit costs have, however, been escalated to year 2017 dollars.

In this concept, the transmuter/energy generation device contains features both of a nuclear fission system (which via nuclear reaction burns fissile and fertile very heavy elements with atomic number > 90) and a nuclear fusion system (which via nuclear reaction burns the very lightest elements with atomic number of 3 or less). Basically the FF (fission/fusion) reactor would consist of a fusion reactor surrounded by blankets consisting a heat removal medium, such as Li-6 enriched lithium metal or a Li-containing molten salt, and fertile fission reactor fuel such as solid or liquid heavy metal forms containing thorium-232 and/or uranium-238 in subcritical configuration.

The concepts studied to date are those based on the deuterium-tritium (D-T) fusion reaction:

$_1H^2 + {}_1H^3 \rightarrow {}_2He^4$ (3.52 MeV) + $_0n^1$ (14.06 MeV)

The energetic 14MeV "fast" neutrons produced are capable of fissioning U-238, which in a thermal neutron system like an LWR, will not fission. Additional "fast" neutrons are also formed by this U-238 fission. Some of the fast neutrons from both fusion and fission can also convert thorium-232 to uranium-233, a fissile uranium isotope that can be burned in thermal reactors, thus establishing an overall nuclear power enterprise symbiosis with thermal fission systems such as LWRs. Useable heat for electricity generation is produced by both the fusion and fission reactors and is removed via the circulating lithium-based coolant. Tritium can be removed from the lithium coolant where it is produced by the (Li-6, n) and (Li-7, n) reactions as follows:

$_3Li^6 + {}_0n^1 \rightarrow {}_1H^3 + {}_2He^4$

$_3Li^7 + {}_0n^1 \rightarrow {}_1H^3 + {}_2He^4 + {}_0n^1$

This "bred" tritium is then recycled to the fusion reactor. Figure RP9-1 shows a diagram of the basic fission/fusion hybrid concept. The following list contains the advantages of this concept per its proponents:

- It makes efficient use of the fertile materials U-238 and Th-232
- The fuel cycle does not require uranium enrichment, a non-proliferation advantage
- The fuel cycle can be operated as an electricity generator, a nuclear waste "incinerator", or a breeder for Pu-239 and/or U-233. Combinations of these functions are possible
- One FF reactor system of a given thermal output can produce fuel for several LWRs of the same thermal output
- The "Q" (ratio of "energy out" to "energy in" for the fusion part of the FF system) can be much lower than for a pure fusion electrical generation reactor
- The nuclear reactor part of the FF system can be subcritical, which may be a safety and operational advantage. This is also true for the Accelerator Driven System, ADS, discussed in Module R6
- The 14 MeV fast neutrons from fusion allow the fission part to act as an "energy amplifier" by the generation of additional fissions and neutrons

- Deuterium is a natural component of the earth's waters, thus the supply is nearly unlimited. Tritium is regenerated from the (Li6,n) reaction, and lithium is also very abundant in the earth's crust.

References RP9-1, RP9-2, and RP9-3 treat Fission/Fusion Hybrids in general and should provide useful background information to the user.

Presently there are two fusion reactor concepts being developed in the US that could potentially serve as the "fusion" part of a hypothetical FF system. Both concepts heat D-T to temperatures of about 10 keV (100,000,000 K), required for a significant fusion cross section and reaction. To fuse a significant fraction of the DT, the product of plasma density and confinement time is the key parameter. Higher density results in more frequent collisions and hence more D-T collisions for a given confinement time, and higher confinement time results in more D-T collisions for a given plasma density.

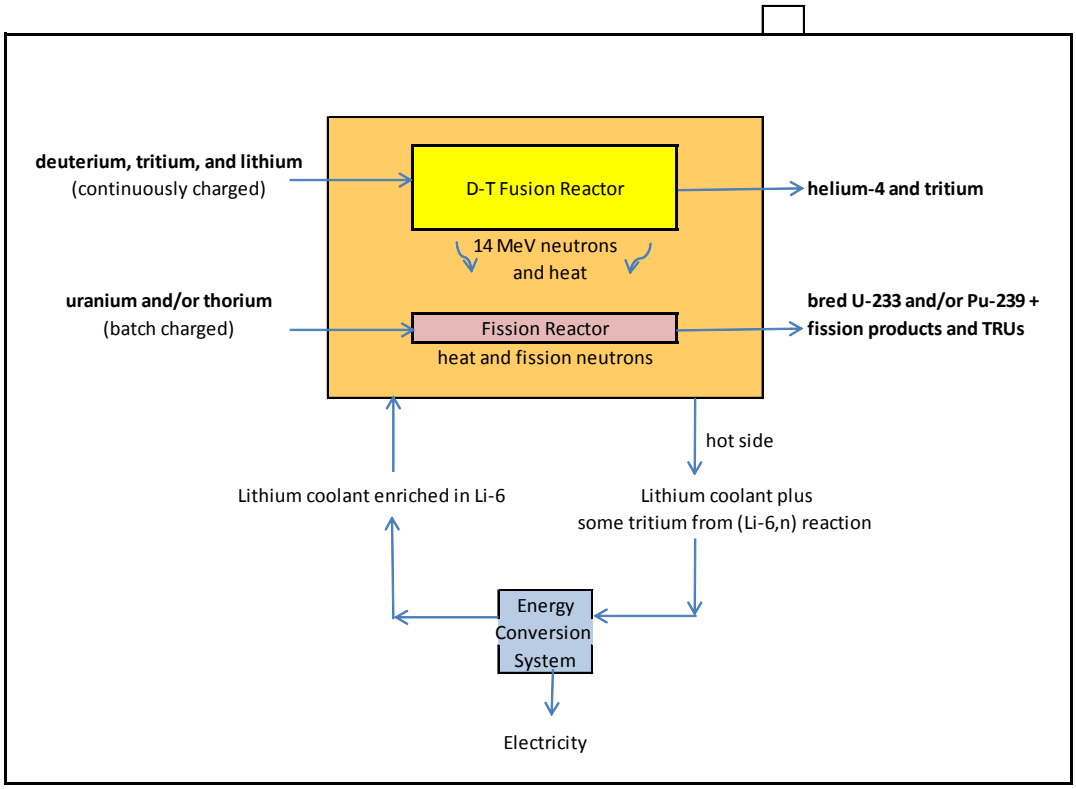

Figure RP9-1. The Generic Fission/Fusion System Concept. (The fission reactor in the figure is a subcritical fission blanket cooled by lithium. Outside fission reactors fueled by the bred U-233 or Pu-239 are not shown).

The magnetic confinement (MC) method uses a very high magnetic field and a high-energy plasma injection scheme to contain a D-T mixture such that a fusion reaction can be sustained for a significant confinement time. The plasma density is limited by the stress limits of the magnets and container (plasma pressure is high because of the high temperature). A lithium blanket would provide heat removal and regeneration of tritium fuel. The international ITER Project, in which the US participates, is constructing a 500 MWth demonstration MC reactor in France that uses the doughnut-shaped "tokamak" configuration for plasma confinement. Other plasma/magnet configurations such as "magnetic mirrors" or "stellarators" are possible for MC fusion.

The inertial confinement method uses a "driver", which directly or indirectly (via x-rays) ablates and implodes a millimeter-scale sphere containing the D-T mixture. Various drivers have been considered including lasers, heavy ion beams and pulsed power (Z-pinch). The confinement time is set by the time it

takes a sound wave to traverse the compressed sphere, and the required density is thus about 100 times that of solid lead. At the time of implosion stagnation, the kinetic energy of implosion is converted to heat, producing the required plasma temperature at the central hot spot of the plasma. In the compressed plasma, the density is high enough to capture the fusion alpha particles in the layer surrounding the central hot spot, heating it to the required temperature. This thermonuclear wave persists for the short time it takes the compressed sphere to disassemble. The overall efficiency of this process can be very high – with fuel burn-up of roughly 30% per shot. After the fusion chamber clears, another sphere is injected and the process repeats; a repetition rate between 5-20 Hz can produce a significant amount of power. Inertial confinement fusion using a laser driver and indirect-drive target is the most advanced. This approach is being pursued in the US by a team led by Lawrence Livermore National Laboratory (LLNL). The National Ignition Facility (NIF) demonstration facility (Ref. RP9-4) is now operating at LLNL for development of this concept. The major purpose of this DOE Defense Programs project is to simulate thermonuclear weapons effects and support the NNSA Stockpile Stewardship program. Pulsed power driven ICF is being led by Sandia National Laboratories, and heavy ion fusion related basic science R&D is led by Lawrence Berkeley National Laboratory.

LLNL is also pursuing a pure fusion power plant design, called Laser Inertial Fusion Energy (LIFE). The LIFE design is derived from the anticipated potential of the NIF to provide full-scale performance demonstration data for a 1000 MWe plant. Figure RP9-2 shows a conceptual LIFE Engine (called so by the LIFE team due to its repetitive mode of operation, distinct from a reactor). LIFE (References RP9-5 and RP9-6) is a pure fusion system with the blanket breeding tritium (for its own use and for use in other systems) and capturing the heat. The fusion reaction is in the small pea-size capsule irradiated by the laser on the fly at the center of the chamber. The chamber shape itself is a transition from a sphere (which is natural given the point source) and a cylinder (which is well suited to a blanket arrangement conducive to use of replaceable modules). Figure RP9-3 from Ref. RP9-5 shows a view of the LIFE system that includes the lasers and the maintenance area for the replacement chambers. The lasers (where the red beams start; the red beams change to blue after frequency conversion) are much smaller than the NIF laser modules, and there is a conceptual design to operate the lasers at the required repetition rate. Figure RP9-4 from Ref. RP9-5 shows a view of the LIFE systems that includes the entire plant footprint.

FFH concepts were studied earlier by LLNL, but are not currently being pursued. See Module R9-2 for more discussion.

Because the fusion concepts are markedly different in configuration and proposed operation, any FF concepts utilizing either MC or IC will be markedly different from the other, Also, different organizations are involved in the research and development and scoping work. The inertial confinement pure fusion configuration is being investigated at LLNL in a small adjunct effort to the NIF work. The magnetic confinement FF concept is mainly a university research project, with most work being conducted at the University of Texas, Austin; Georgia Tech; and the University of California at San Diego. LLNL has also conducted some paper studies on the MC method using magnetic mirrors (References RP9-7, RP9-8, and R9-P.9). For this reason, the authors have split this R9 Module into two parts: Module R9-1 deals with the magnetic confinement FF method, and Module R9-2 with the inertial confinement FF concept.

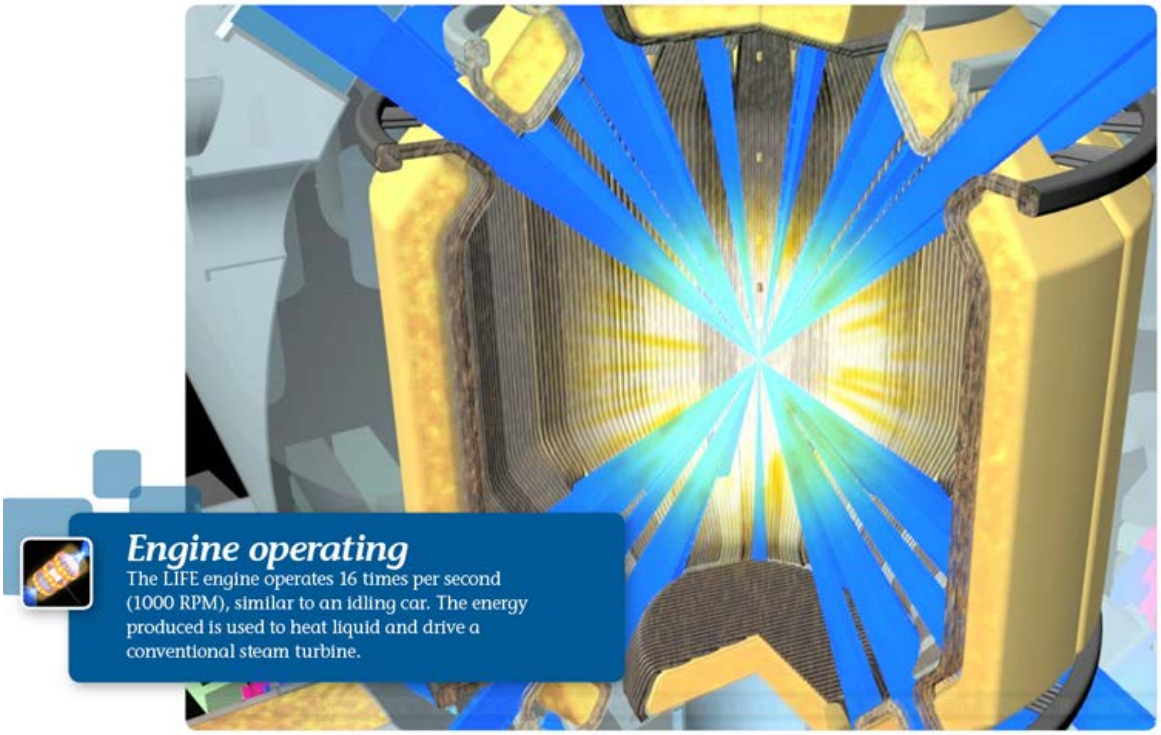

Figure RP9-2. Conceptual Scheme for the LIFE IC System Pure Fusion Chamber.

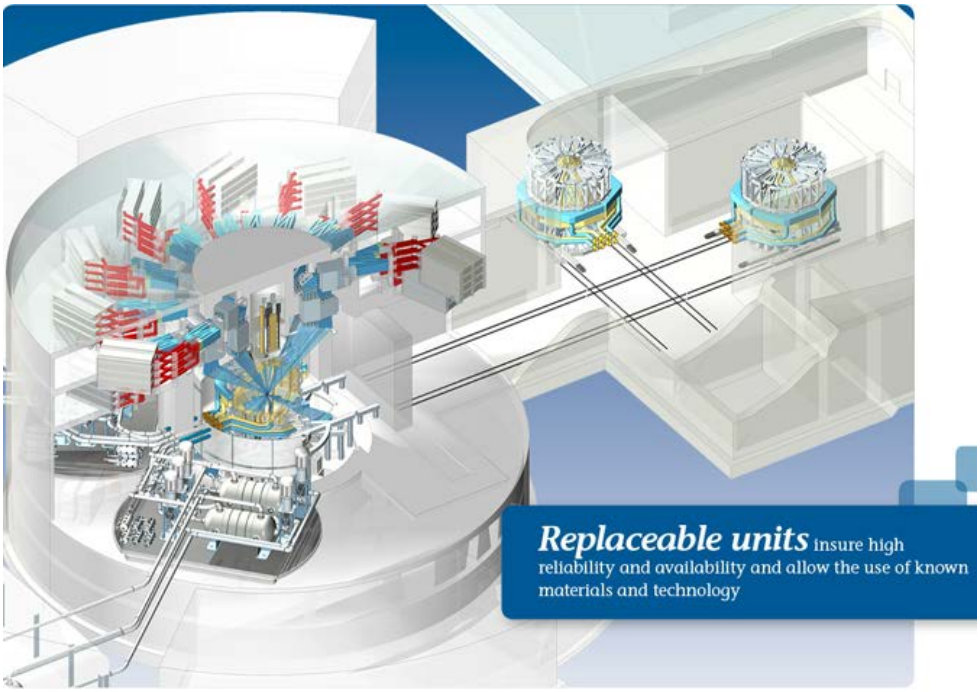

Figure RP9-3. The LIFE IC System Lasers and Maintenance Area for a pure fusion system.

Figure RP9-4. The LIFE IC System Overall Plant Configuration for a pure fusion system.

Figure RP9-5. Cutaway View of ITER pure fusion reactor.

Figure RP9-5 above shows a cutaway schematic of the ITER pure fusion magnetic confinement reactor under construction in France. This can be compared to the IC pure fusion schematics preceding it.

RP9-2. REFERENCES/BIBLIOGRAPHY

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RP9-7 Moir, R.W., et al.; *Design of Helium-cooled Molten Salt Fusion Breeder*; **Fusion Technology**; Vol 8, 1985, pp 465-473

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Module R9-1

Fission/Fusion Hybrid Systems: Magnetic Confinement D-T Fusion

Module R9-1

Fission/Fusion Hybrid Systems: Magnetic Confinement D-T Fusion

R9-1.MD. SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2012 AFC-CBR)
- **Estimating Methodology for latest (2012 AFC-CBR) technical update from which this 2017 update was escalated:** System descriptions from fusion research laboratories and universities were utilized, along with some scaling calculations, to develop rough cost estimates. These were combined with engineering judgment of the overall system differences from LWRs.

R9-1.RH. REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2012 as Module R9-1.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2012
- **New technical/cost data which has recently become available and will benefit next revision:**
  - The use of this unit cost information (and its format) in the Evaluation and Screening Report [Reference R9-1.9] did not mesh well with the range of FFH concepts considered in the screening exercise. The cost of the MC fusion reactor needs to be cast in terms of the size of the fusion device required, which is in turn a function of the degree of sub-criticality and thermal power of the reactor. In addition, the electricity requirements or net production should be included as an annual cost (or revenue) like O&M. Essentially, instead of rolling all the system (fusion reactor +subcritical reactor) cost assumptions into a final $/kWe, they should be provided separately so the cost estimators can more accurately assess life cycle costs for the specific FFH system that they are evaluating.

R9-1-1. BASIC INFORMATION

This module has been newly drafted for the FY2012 Cost Basis Report update. It is concerned with the capital cost of Magnetic Confinement (MC) Hybrid Fission/Fusion Systems (FF), defined in this module as industrial scale Nth-of-a-kind (NOAK) electricity production machines. As with ADS (Module R6), it is highlighted that no such machine has been constructed nor operated as of yet; therefore, all the costs presented here are derived from paper studies based on hypothetical systems. The capital cost is best subdivided between the fusion and fission parts of the overall structure. This is similar to how the system cost for ADS was partitioned between the accelerator and subcritical reactor portions. In particular, most of the cost data on the MC fusion part of the FF system are derived from studies related to ARIES, ITER, and other pre-conceptual design magnetic confinement fusion reactor (MCFR) studies. The International ITER (Latin for "the way" *to future energy*) fusion reactor project is advancing towards demonstration of the MCF technology, with early construction and equipment procurement underway.

The capital costs for the fission reactor portion of an FFH system are based on those for relevant reactor systems covered in earlier R-modules, particularly the subcritical fission system for ADS (Module

R6) and the critical fast reactor (Module R2). Most of this section deals with costs for the MC fusion reactor. The MCFR for the FFH system is likely to be a Tokamak-based system, such as ITER and the one proposed by the University of Texas, Austin (Ref. R9-1.1), or a magnetic mirror concept (Ref. R9-1.2).

R9-1-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

Magnetic Confinement Hybrid Systems (FFHSs) are composed of two coupled main parts: (1) a sub-critical fission blanket (reactor) in which the power is maintained at the desired level through the use of an external neutron source and (2) a magnetic confinement fusion reactor (MCFR) that, through the use of the D-T fusion reaction, generates the fast 14 MeV source neutrons capable of fissioning U-238, sustaining fission, and/or breeding Pu-239 from U-238 or U-233 from Th-232.

It is assumed that, by dropping the constraint of maintaining the nuclear core criticality, additional flexibility can be gained by MC-FFHs as opposed to critical fission transmuters. This flexibility in turn can be used for certain types of reactor applications, such as actinide burning (transmutation of large quantities of heavier minor actinides – MAs) or breeding of Pu-239 or U-233 for use in fission-only reactors (sustainable nuclear power via symbiosis). In fact, there are not substantial technological differences between fast reactors and the subcritical parts of the FFH system; therefore, the cost of a fast reactor (FR, Module R2) is assumed as one of the bases for the cost of the subcritical part of the FFHS (and ADS).

R9-1-3. PICTURES AND DIAGRAMS

Figure R9-1.1 shows a conceptual scheme for an MC-FFH system. This example, from the University of Texas Program (Ref. R9-1.1) is a waste burner. The “Fission Waste” section shown in the diagram could be fission heat source for power generation and/or a breeder of fissile materials from fertile U-238 or Th-232. The fusion reactor is of the familiar donut-shaped tokamak design.

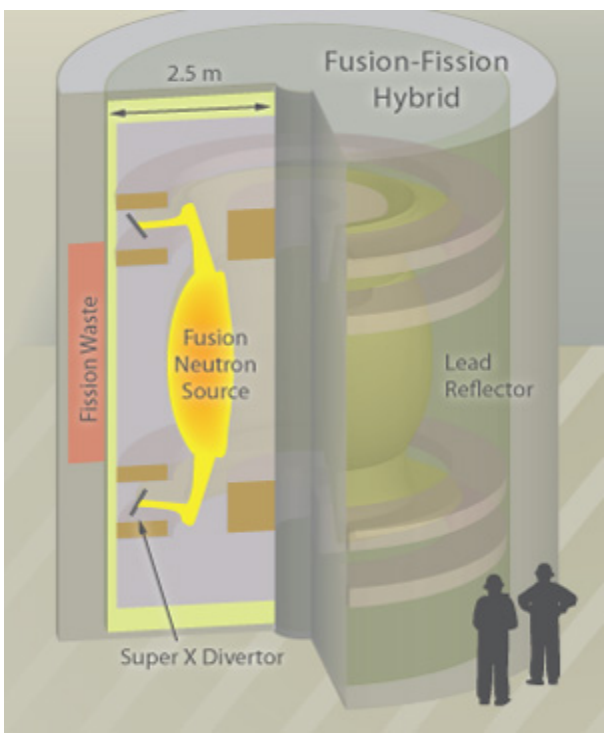

Figure R9-1.1. Conceptual Scheme for the MC-FFH System.

R9-1-4. MODULE INTERFACES

The subcritical fission portion of the MC-FFH system receives solid fertile fuel assemblies from the fuel fabrication plant, which can be central or co-located with the MC-FFH facility. The liquid coolant, most likely lithium metal or a molten salt lithium compound, removes useable heat from both the fusion and fission portions of the blanket surrounding the fusion source and transports it to the power conversion system for electricity generation. After irradiation, the discharged highly radioactive fission fuel would be kept in wet-storage on site until ready for on-site storage or off-site storage or disposal (for the actinide burner application), or reprocessing and recovery of useful fissile materials (for the sustainable nuclear power by symbiosis application). Although both aqueous and electrochemical reprocessing are possible, electro-refining is often envisioned as the fuel reprocessing method for the discharged MC-FFH metal fuel. These integral fuel recycle facilities would be normally co-located with the reactor systems. Co-location would save the off-site transportation costs and enhance security. It should be noted that there is also a FFH concept that marries the features of the liquid-fueled molten salt reactor (Module R7) and fusion reactors.

R9-1-5. SCALING CONSIDERATIONS

Both the subcritical fission system and the MCFR system should feature a reduction in specific overnight costs ($/kW) with increasing system power for each.

R9-1-6. COST BASES, ASSUMPTION AND DATA SOURCES

The cost summary is divided in two parts: (1) the MC fusion reactor (capital and O&M) and (2) the subcritical blanket and the power conversion equipment (capital and O&M). In this section, any fuel processing/fabrication facilities are excluded, since they are treated in detail in the D and F modules. In order to cost out the whole FFH transmutation system, total neutron and power balances are required. For the neutron balance, one begins with the neutrons required to breed tritium. Most of the tritium to be bred is to replace tritium consumed in fusion reactions, but a small additional amount is needed to make up for losses in the fusion chamber exhaust and tritium processing systems. The remaining neutrons are available for use in the subcritical fission blanket surrounding the fusion chamber. Those neutrons can be absorbed to breed U-233 or Pu-239, fast fission U-238, or can be absorbed by other materials or leak through the neutron reflector. The amount of fast fissions per fusion reaction (and also the consideration of energy released or consumed by non-fission nuclear reactions in the blanket) determines the balance of power between the fusion and fission portions of the FFH system. To roughly calculate the overall system specific overnight capital cost one can use the following relationship:

$/kW for total FFH machine =

(% of net power from fusion reactor portion) x ($/kW for MCFR) +
(% of net power from fission reactor portion) x ($/kW for fission reactor)

A key parameter for the fusion system is the “Q” (ratio of fusion power out to the “power in” required to drive the magnets and injectors which sustain the fusion reaction) of the MCFR. For a FFH system, one has additional cost for the fission blanket, but also has additional energy from the fission reactions. This means that a lower “Q” is required for the fusion reactor in order to provide a given amount of overall FFH system power. Bethe, in Reference RP9-1, showed example energy balances.

The nature of the fission reactor fuel influences both the neutron and the power balances. For a uranium system, U-238 fissions are more likely than neutron absorption and transmutation for a flux of 14 MeV neutrons, and the fission component of the FFH can produce a higher percentage of the heat eventually converted to electricity. In contrast, for a thorium system, a higher percentage of the 14 MeV neutrons from fusion would be used to convert thorium to U-233 by absorption (without the energy

release from fission in the FFH system). A higher percentage of the useable heat for electricity production by the FFH system would come from the fusion portion of the FFY.

For rough life cycle estimating, the system O&M costs, expressed in $/kWe-y, could be calculated in the same manner using power partitioning.

Capital and O&M Cost of the Subcritical Blanket (Reactor). The basic assumption in the ADS (Module R6) and FFH cost studies is that the capital cost of the sub-critical part of the system will be similar – but slightly higher – to that of a critical fast reactor similar in size/power level to the sub-critical unit. As with the ADS, the specific cost (i.e. $/kWe) of the reactor/power conversion portion of the facility will be higher than that of a similar critical fast reactor because:

- The extra size of the plant necessary to generate part of the electricity needed to run its own pumps and part of the fusion reactor input power system (magnets, tritium recovery, injectors, etc.); this is electricity that is not available for sale. The extra electricity needed is about 8% of the total in the case of ATW; a similar or higher number is assumed for the FFH system depending on the "Q" of the MCFR (10% higher is assumed for this case). In addition to the standard FR components, there will be extra complications such as coolant/blanket connections for both nuclear fuel rods and tritium removal subsystems.
- Some components will be absent or reduced, such as reactor control rods, but the cost benefit of this is likely to be over-compensated by the extra cost of components needed in FFH systems and not in a FR.
- The subcritical blanket geometry will be subject to the constraints due to the toroidal plasma vessel and the surrounding magnets.

The Advanced Liquid Metal Reactor (ALMR) has been used as a reasonable cost basis for ATW because of the large amount of work done on the cost of the ALMR (funded by DOE from 1989 to 1995) as documented in Module R2. Table R9-1.1 gives the specific capital cost for the critical fast reactor (Module R2) and the subcritical portion of the FFH system including its share of the overall steam generator and turbine costs. This range is close to but slightly above that used for the sodium-cooled fast critical reactor in Module R2 and encompasses the range of subcritical reactor costs discussed in Module R6 (ADS) as well as the overnight cost of $4,000/kWe in the 2009 update to the MIT Future of Nuclear Power study. Note that for the ATW system discussed in Module R6, there was a more-detailed pre-conceptual design and cost estimate available from the ATW Program that could be analyzed.

The range for the fixed component of O&M cost was obtained in a similar fashion. The low end of the range was taken from Module R2, for a critical FR. The high end of the range was taken from Module R6, for an Accelerator Driven System.

Table R9-1.1 What-it-Takes Cost Range for Critical Fast Reactors and the Subcritical Portion of a FFH System (2012 $).

Item	Low Cost	High Cost
Subcritical Reactor Portion of the FFH system (Specific Overnight Cost)	$2,100/kWe NOAK (10% above critical FR due to complexity)	$6,600/kWe NOAK (10% above critical FR due to complexity)
Subcritical Reactor Portion of the FFH system (Fixed Component of O&M costs)	$60/kWe-y (same as critical FR)	$230/kWe-y (same as ADS Module R6)
Critical Fast Reactor (Specific Overnight Cost) Module R2	$1,900/kWe NOAK (from Module R2)	$6,000/kWe NOAK (from Module R2)
Critical Fast Reactor (Fixed Component of O&M costs)	$60/kWe-y (from Module R2)	$85/kWe-y (from Module R2)

Capital and O&M Costs of the MC Fusion Reactor. Over the 60+ years that fusion energy has been pursued, there have been numerous cost estimates and cost models developed for MCFR concepts. None are very recent (last 5 years) and do a credible NOAK-to-NOAK comparison against a LWR reactor. Descriptions of models such as SYMECON (Ref. R9-1.5) were found, but the model itself was not available. There is a cost estimate for the ITER design, but ITER is a developmental project not wholly representative of a commercial MCFR design. Table R9-1.2 below shows a compilation of cost information gleaned from the literature and internet sources. Most values given were for the MCFR only, but a few cost estimates for complete FFH systems were also found and are presented. Most values have been converted to specific overnight cost ($/kWe) and to 2012 constant dollars. In some cases, O&M costs were available, but only for the fixed ($/kWe-y) component. This is still useful, since for most reactor systems, the fixed component is considerably larger than the variable component (which varies with power production).

It is not known what contingency values were added to the base construction cost in the numbers given above. At this point in MCFR development, anything lower than 50% is probably not realistic. (This would be the cost contingency only. The possibility that the fusion system will not achieve its stated design performance in terms of net power output and capacity factor is not considered here, as successful development of the concept must be assumed to provide the basis for NOAK costs. However, successful development is a valid concern given the low technology readiness level of the concept and the significant technical issues remaining.

Table R9-1.2 Cost data on Magnetic Confinement Fusion Reactors from Literature Sources.

Reference Item	Cost Data
Bethe (Ref. RP9-1) [1979]	For a fusion reactor with "Q" of 10, specific MCFR capital cost is 3 times that of LWR (multiplying Module R1 range by 3 gives $6,900/kWe to $17,400/kWe). Whole FFH system specific capital cost is 1.26 to 1.46 times that of a LWR (depends on MCFR "Q" and fuels used in fission reactor, the range would be $2,100/kWe to $8,500/kWe). R&D would cost $10B in 1979$ ($35B in 2012$).
Beyond ITER (Ref. R9-1.4) [2008]	1.5 GWe FOAK MCFR = $8,000/kWe (2012$) 1.5 GWe NOAK MCFR = $4,000/kWe (2012$) Article uses $3,000/kWe for NOAK LWR fission.
2011 Update of Technology Map (Ref. R9-1.3)	Large DEMO follow on to ITER will cost $10,000/kWe (1995$); $14,500/kWe in 2012$
Moir (Ref. 9-1.2) [2012]	1.38 GWe FFH System would cost $4.87B in 1982$. Would be ~15B in 2012$
UCSD (Ref. 9-1.5) [2008]	States consensus view that $/kWe for MCFR will be greater than for LWRs
ORNL (Ref. R9-1.8) [2000]	Overnight NOAK range is $4,000 to $5,700/kWe in 1999$. ($5,600/kWe to $8,000/kWe in 2012$). Report assumed ARIES type tokamak MCFR was used as base case. Fixed O&M cost of $60/kWe-y assumed ($84/kWe-y in 2012$).

R9-1-7. DATA LIMITATIONS

No FFH system or continuously-operating MCFR has been constructed and operated to date; therefore, the cost assumptions presented here are largely estimates of costs based on paper studies for hypothetical systems. In fact, most of the MCFR data in this section rely on cost projections made for MCFRs that would follow the 500 MWe ITER MCFR being constructed in France. These studies (Refs. 9-1.3 and 9.1-4) have tried to project NOAK MCFR costs based on ITER and follow-on demonstration costs. (ITER's project cost estimate started around $5B, but is now estimated to be closer to $15B.) The cost uncertainties for MCFRs are much higher than for fission reactors of any type. This is a result of the

much lower technical maturity of controlled fusion vis-à-vis controlled fission. (Fusion reactors have not yet reached the goal of sustained energy "breakeven" output that the first fission reactors reached during the early Manhattan project [1942]). Ref. R9-1.5 discusses the very considerable R&D needs for commercial fusion. Ref. R9-1.6 discusses the R&D needs for FFH systems. Ref. R9-1.7 presents the views of skeptics of FFH technology.

R9-1-8. COST SUMMARIES

The specific costs of the two major components of a MC FFH system are summarized in the "What-it-Takes" Table R9-1.3. In order to calculate the specific costs for a complete FFH system, the user (fuel cycle analyst in this case) must define the mission, material balance, and energy balance for the whole system. (The formula for apportioning cost by power output is discussed in Section R9-1-6.)

The operation and capital costs of both the subcritical blanket (reactor) and MCFR portions of the FFH are higher than those of most critical reactors, mostly because of the added costs of the fusion reactor system and its integration with the fission portion, and also because of the higher complexity and integration requirements of the subcritical system. The technical maturity of this system compared to the ADS is smaller, because accelerators have been built on a fairly large scale and are successfully operating (e.g. the Spallation Neutron Source in Oak Ridge).

For the NOAK case, the subcritical reactor part of the capital cost range has been estimated at 2100 $/kWe, to $6600/kWe by adding 10% to both the ALMR (Module R2) low and high "What-it-Takes" specific costs. The low end of the MCFR range has been derived by escalating the low-end specific overnight cost value from the 1999 Oak Ridge (Delene, et al.) cost study (Ref. R9-1.8). The high-end value is triple the low-end value. A factor of three was used to account for the very low technical maturity, high design uncertainties (including the present non-availability of very radiation and heat resistant materials), and tripled estimated cost of the ITER Project. Nominal values were selected near the midpoints of these ranges.

The "What-it-Takes" fixed component of the O&M cost was derived in a similar manner, with the ORNL report (Ref R9-1.8) providing a low value (after escalation) and a doubling of this value for the high O&M cost.

Table R9-1.3. What-It-Takes Cost Summary Table for a MC FFH System (from 2012 AFC-CBD Update; Year 2012$.

	Upside (Low Cost)	Nominal (most likely cost)	Downside (High Cost)
Capital Cost of the Subcritical Reactor	2,100 ($/kWe)	4,400 ($/kWe)	6,600 ($/kWe)
Capital Cost of the MC Fusion Reactor	5,600 ($/kWe)	11,000 ($/kWe)	16,000 ($/kWe)
O&M Cost of the Subcritical Reactor	60 ($/kWe-y)	100 ($/kWe-y)	230 ($/kWe-y)
O&M Cost of the MC Fusion Reactor	80 ($/kWe-y)	120 ($/kWe-y)	160 ($/kWe-y)

The following Table R9-1.4 provides the same data escalated to Year 2017$ (1.088 escalation factor followed by rounding).

Table R9-1.4. What-It-Takes Cost Summary Table for a MC FFH System in Year 2017$.

	Low Cost	High Cost	Mean Cost	Mode Cost
Capital Cost of the Subcritical Reactor	2,300 ($/kWe)	7,200 ($/kWe)	4,800 ($/kWe)	4,800 ($/kWe)
Capital Cost of the MC Fusion Reactor	6,100 ($/kWe)	17,400 ($/kWe)	11,800 ($/kWe)	12,000 ($/kWe)
O&M Cost of the Subcritical Reactor	65 ($/kWe-y)	250 ($/kWe-y)	141 ($/kWe-y)	109 ($/kWe-y)
O&M Cost of the MC Fusion Reactor	87 ($/kWe-y)	174 ($/kWe-y)	131 ($/kWe-y)	131 ($/kWe-y)

Figure F9-1.2 shows the cost-related probability distributions for the Magnetic Confinement variant of the Fission/Fusion hybrid Option.

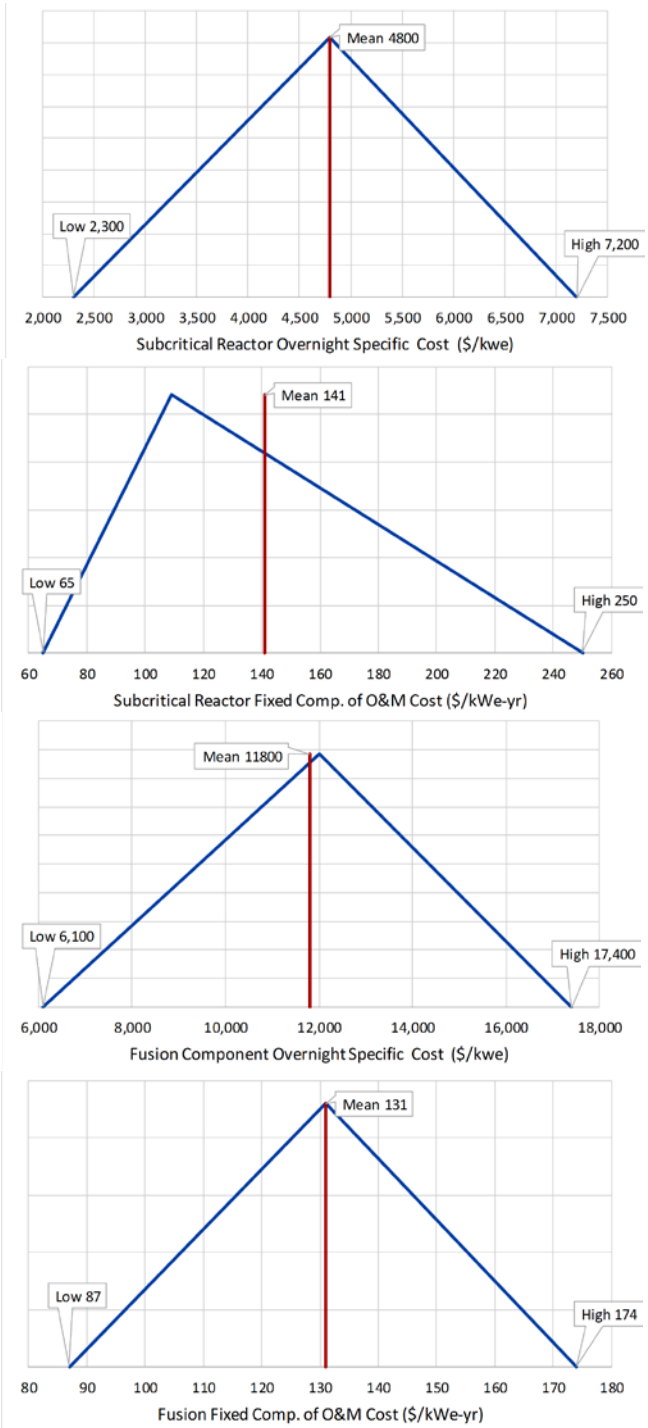

Figure R9-1.2. Probability Distributions for Hybrid Fission/Fusion System: Magnetic Confinement Fusion.

R9-1-9. SENSITIVITY ANALYSES

None available.

R9-1-10. REFERENCES

R9-1.1 Institute for Fusion Studies, University of Texas, Austin; Nuclear Fusion-Fission Hybrid Could Destroy Nuclear Waste and Contribute to Carbon-Free Energy Future; Jan 27, 2009

R9-1.2 Moir, R.W., et al.; Axisymmetric Magnetic Mirror Fusion-Fission Hybrid; Transactions of Fusion Science and Technology; Vol 61; Jan 2012

R9-1.3 Section 11: Update of the Technology Map for the SET-Plan: Nuclear Fusion Power Generation; pages 83-88

R9-1.4 Beyond ITER; from www.iter.org/Future-beyond.htm

R9-1.5 Tillack, M.S., et al.; Issues and R&D needs for commercial fusion energy: An interim report of the ARIES technical working groups; Center for Energy Research; University of California, San Diego; UCSD-CER-08-01; July 2008

R9-1.6 USDOE; Research Needs for Fusion-Fission Hybrid Systems; Report of the Research Needs Workshop (ReNeW); USDOE Workshop: Sept 30-Oct 2, 2009; Gaithersburg, MD

R9-1.7 Afeyan, B. et al.; A Skeptical Assessment of Fission-Fusion Hybrids (V1a); viewgraph presentation from the Workshop cited in R9-1.6 above)

R9-1.8 Delene, J.G., Williams, K.A., et al.; An Assessment of the Economics of Future Electric Power Generation Options and the Implications for Fusion – Revision 1; ORNL/TM-1999/243/R1; Oak Ridge National Laboratory; January 2000

R9-1.9 R. Wigeland, T. Taiwo, H. Ludewig, M. Todosow, W. Halsey, J. Gehin, R. Jubin, J. Buelt, S. Stockinger, K. Jenni, and B. Oakley, Nuclear Fuel Cycle Evaluation and Screening – Final Report, FCRD-FCO-2014-000106, INL/EXT-14-31465, Idaho National Laboratory, October 7, 2014.

Module R9-2

Fission/Fusion Hybrid Systems: Inertial Confinement D-T Fusion

Module R9-2

Fission/Fusion Hybrid Systems: Inertial Confinement D-T Fusion

R9-2.MD SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year for 2017 Update:** FY 2017
- **Nature of this 2017 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2012 AFC-CBR)
- **Estimating Methodology for latest (2012 AFC-CBR) technical update from which this 2017 update was escalated:** Technology descriptions from LLNL along with scaling relationships were utilized to develop a rough lifecycle cost estimate for a Fission-Fusion Hybrid system. The analysts also took into account engineering judgment of the differences of this FFH system from LWRs.

R9-2.RH REVISION HISTORY

- **Version of AFC-CBR in which Module first appeared:** 2012 as Module R7.
- **Latest version of module in which new technical data was used to establish unit cost ranges:** 2012
- **New technical/cost data which has recently become available and will benefit next revision:**
  - The use of this unit cost formatted AFC-CBR information in the Evaluation and Screening Report [Reference R9-2.14] did not mesh well with the range of FFH concepts considered in the screening exercise. The cost of the fusion reactor needs to be cast in terms of the size of the fusion device required, which is in turn a function of the degree of sub-criticality and thermal power of the fission reactor. In addition, the electricity requirements or net production should be included as a cost (or revenue) like O&M. Essentially, instead of rolling all the cost-related assumptions into a final $/kWe, they should be provided separately so the cost estimator can estimate more accurately the specific FFH system life cycle costs they are evaluating.

R9-2.1. BASIC INFORMATION

This module has been newly drafted for the FY2012 Cost Basis Report update. It is concerned with the capital cost of Inertial Confinement (IC) Hybrid Fission/Fusion Systems (FF), defined in this module as industrial scale Nth-of-a-kind (NOAK) electricity production machines. As with ADS (Module R6), it is highlighted that no such machine has been constructed nor operated as of yet; therefore, all the costs presented here are derived from paper studies based on hypothetical systems. Furthermore, the most credible cost estimate for a pure fusion IC system should be used with caution when applying it to a FF system, because the FF system will likely be subject to NQA-1 quality assurance and Nuclear Regulatory Commission licensing processes, whereas a pure fusion system may be regulated under a system that can be exercised with less cost and lead time. The pure fusion cost estimates in this section assume the simpler regulatory system.

The capital cost is most easily subdivided between the fusion and fission parts of the overall structure, although of course this does not provide a rigorous cost basis for an integrated plant. This is similar to how the system cost for ADS was partitioned between the accelerator and subcritical reactor portions. In

particular, most of the cost data on the IC fusion part of the FF system are derived from pure fusion inertial confinement fusion studies, which rely on data from the National Ignition Facility (NIF) and industrial consultations drawn from the LIFE project. The NIF project is advancing towards demonstration of the Inertial Confinement Fusion (ICF) technology, with construction complete, the initial National Ignition Campaign complete, and with ongoing calculations and experiments working toward demonstration of ignition (net energy gain).

The capital costs for the fission reactor portion of an FFH system are based on those for relevant reactor systems covered in earlier R-modules, particularly the subcritical fission system for ADS (Module R6) and the critical fast reactor (Module R2). Most of this section deals with costs for the IC fusion plant. The pure fusion system is likely to be a solid-state laser-based system, using technology developed for various industrial applications (such as semiconductor laser diodes), coupled to information arising from the NIF project. Ongoing research at LLNL is addressing reduction in physical size of the laser driver, removal of laser heat to accommodate a 5-20 Hz repetition rate, and mass production of the fusion fuel (References R9-2.1, R9-2.2 and R9-2.3). It is assumed that the pure fusion design can be the starting point for a FF hybrid system design, if that concept is pursued in the future, with the substantial caveat that the licensing regime and overall system design may be quite different for the FF system.

R9-2.2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

Inertial Confinement Hybrid Systems (FFHSs) are composed of two coupled main parts: (1) a sub-critical fission blanket (reactor) in which the power is maintained at the desired level through the use of an external neutron source and (2) an inertial confinement fusion system that, through the use of the D-T fusion reaction, generates the fast 14 MeV source neutrons capable of fissioning U-238, sustaining fission, and/or breeding Pu-239 from U-238 or U-233 from Th-232.

It is assumed that, by dropping the constraint of maintaining the nuclear core criticality, additional flexibility can be gained by IC-FFHs as opposed to critical fission transmuters. This flexibility in turn can be used for certain types of reactor applications, such as actinide burning (transmutation of large quantities of heavier minor actinides – MA) or breeding of Pu-239 or U-233 for use in fission-only reactors (sustainable nuclear power via symbiosis). In fact, at a conceptual level there are not substantial technological differences between fast reactors and the subcritical parts of the FFH system; therefore, the cost of a fast reactor (FR, Module R2) is assumed as one of the bases for the cost of the subcritical part of the FFHS (and ADS). Only detailed design activity will determine whether these assumptions are valid or not.

R9-2.3. PICTURES AND DIAGRAMS

The major differences between a FF hybrid and a pure fusion system are a completely different blanket design that can include the fertile materials; the need for a fission fuel processing area; the likely need for substantially more safety structures, systems and components; a different approach to maintenance and replacement of the blanket; and the need for fuel preparation and disposal infrastructure.. The licensing regime for a FF hybrid will be much more stringent than for a pure fusion system, with criticality control and decay heat playing a key role in the design; hence cost estimates for the pure fusion system should be used with caution. References R9-2.4, R9-2.5, and R9-2.6 discuss prior FF hybrid designs developed at LLNL in more detail; current LLNL studies are focusing on the pure fusion option.

Figure R9-2.1 from Reference R9-2.4 shows the fusion chamber and subcritical blanket for a system producing 375-500 MW of fusion power and a total of 2000-5000 of total thermal power.

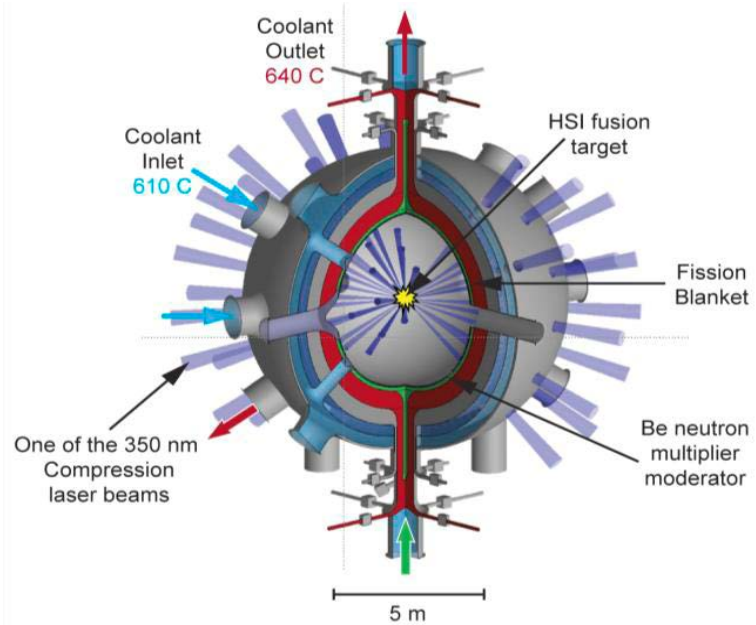

Figure R9-2.1. Conceptual Scheme for an IC-FFH System.

Figure R9-2.2 from Reference R9-2.7 shows the same system, with more information on the flow of coolant and fuel pebbles.

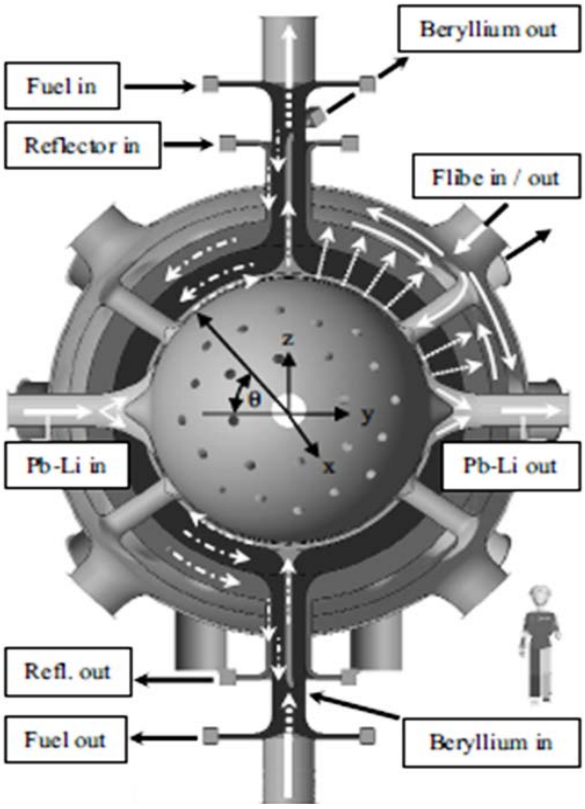

Figure R9-2.2. The internal structures, pebble movements, and coolant flows of an IC FF Hybrid.
Solid arrows: Coolant flow (Pb-Li or Flibe)
Dotted arrows: Coolant flow through pebbles
Dot-Dash arrows: Creeping pebble movement (Be, fuel, reflector)

R9-2-4. MODULE INTERFACES

The subcritical fission portion of the IC-FFH system receives solid fertile fuel assemblies from the fuel fabrication plant, which can be central or co-located with the IC-FFH facility. The liquid coolant removes useable heat from both the fusion and fission portions of the blanket surrounding the fusion source and transports it to the power conversion system for electricity generation. Some significant fraction of this coolant will need to contain lithium, to provide adequate tritium breeding capability. After irradiation, the discharged highly radioactive fission fuel would be kept in wet-storage on site until ready for on-site storage or off-site storage or disposal (for the actinide burner application), or reprocessing and recovery of useful fissile materials (for the sustainable nuclear power by symbiosis application). Although both aqueous and electrochemical reprocessing are possible, electro-refining is often envisioned as the fuel reprocessing method for the discharged IC-FFH metal fuel. These integral fuel recycle facilities would be normally co-located with the reactor systems. Co-location would save the off-site transportation costs and enhance security. It should be noted that there is also a FFH concept that marries the features of the molten salt reactor (Module R7) and fusion reactors.

R9-2-5. SCALING CONSIDERATIONS

Both the subcritical fission system and the ICFR system should feature a reduction in specific overnight costs ($/kW) with increasing system power for each.

R9-2-6. COST BASES, ASSUMPTION AND DATA SOURCES

The cost summary is divided in two parts: (1) the IC fusion plant (capital and O&M) and (2) the subcritical blanket and the power conversion equipment (capital and O&M). In this section, any fuel processing/fabrication facilities are excluded, since they are treated in detail in the D and F modules. In order to cost out the whole FFH transmutation system, total neutron and power balances are required. For the neutron balance, one begins with the neutrons required to breed tritium. Most of the tritium to be bred is to replace tritium consumed in fusion reactions, but a small additional amount is needed to make up for losses in the fusion chamber exhaust and tritium processing systems. The remaining neutrons are available for use in the subcritical fission blanket surrounding the fusion chamber. Those neutrons can be absorbed to breed U-233 or Pu-239, fast fission U-238, or can be absorbed by other materials or leak through the neutron reflector. The amount of fast fissions per fusion reaction (and also the consideration of energy released or consumed by non-fission nuclear reactions in the blanket) determines the balance of power between the fusion and fission portions of the FFH system. To roughly calculate the overall-system specific overnight capital cost one can use the following relationship:

$/kW for total FFH machine =

(% of net power from fusion reactor portion) x ($/kW for fusion plant) +
(% of net power from fission reactor portion) x ($/kW for fission reactor)

A key parameter for the fusion system is the “gain” (ratio of fusion power out to the “power in” required to drive the lasers which sustain the fusion reaction) of the fusion plant. For a FFH system, one has additional cost for the fission blanket, but also has additional energy from the fission reactions. This means that a lower “gain” is required for the fusion plant in order to provide a given amount of overall FFH system power. Bethe, in Reference RP9-1, showed example energy balances.

The nature of the fission reactor fuel influences both the neutron and the power balances. For a uranium system, U-238 fissions are more likely than neutron absorption and transmutation for a flux of 14 MeV neutrons, and the fission component of the FFH can produce a higher percentage of the heat eventually converted to electricity. In contrast, for a thorium system, a higher percentage of the 14 MeV neutrons from fusion would be used to convert thorium to U-233 by absorption (without the energy release from fission in the FFH system). A higher percentage of the useable heat for electricity production by the FFH system would come from the fusion portion of the FFY.

For rough life cycle estimating, the system O&M costs, expressed in $/kWe-y, could be calculated in the same manner using power partitioning.

Capital and O&M Cost of the Subcritical Blanket (Reactor). The basic assumption in the ADS (Module R6) and FFH cost studies is that the capital cost of the sub-critical part of the system will be similar – but slightly higher – to that of a critical fast reactor similar in size/power level to the sub-critical unit. As with the ADS, the specific cost (i.e. $/kWe) of the reactor/power conversion portion of the facility will be higher than that of a similar critical fast reactor because:

1. The extra size of the plant necessary to generate part of the electricity needed to run its own pumps and part of the fusion plant input power system (lasers, tritium recovery, etc.); this is electricity that is not available for sale. The extra electricity needed is about 8% of the total in the case of ATW; a similar or higher number is assumed for the FFH system depending on the "gain" of the fusion plant. In addition to the standard FR components, there will be extra complications such as coolant/blanket connections for both nuclear fuel rods and tritium removal subsystems.
2. Some components will be absent or reduced, such as reactor control rods, but the cost benefit of this is likely to be over-compensated by the extra cost of components needed in FFH systems and not in a FR.
3. The subcritical blanket geometry will be subject to the constraints due to the fusion chamber and laser entrance ports. These constraints are expected to be less significant than the analogous constraints for a MCFR.

The Advanced Liquid Metal Reactor (ALMR) has been used as the cost basis for ATW because of the large amount of work done on the cost of the ALMR (funded by DOE from 1989 to 1995) as documented in Module R2. Table R9-2.1 gives the specific capital cost for the critical fast reactor (Module R2) and the subcritical portion of the FFH system including its share of the overall steam generator and turbine costs. This range is close to but slightly above that used for the sodium-cooled fast critical reactor in Module R2 and encompasses the range of subcritical reactor costs discussed in Module R6 (ADS) as well as the overnight cost of $4000/kWe in the 2009 update to the MIT Future of Nuclear Power study (Ref. R9-2.8). Note that for the ATW system discussed in Module R6, there was a more-detailed pre-conceptual design and cost estimate available from the ATW Program that could be analyzed.

The range for the fixed component of O&M cost was obtained in a similar fashion. The low end of the range was taken from Module R2, for a critical FR. The high end of the range was taken from Module R6 (an ADS).

Table R9-2.1 What-it-Takes Cost Range for Critical Fast Reactors and the Subcritical Portion of a FFH System (2012$).

Item	Low Cost	High Cost
Subcritical Reactor Portion of the FFH system (Specific Overnight Cost)	$2,100/kWe NOAK (10% above critical FR due to complexity)	$6,600/kWe NOAK (10% above critical FR due to complexity)
Subcritical Reactor Portion of the FFH system (Fixed Component of O&M costs)	$60/kWe-y (same as critical FR)	$230/kWe-y (same as ADS)
Critical Fast Reactor (Specific Overnight Cost) Module R2	$1,900/kWe NOAK (from Module R2)	$6,000/kWe NOAK (from Module R2)
Critical Fast Reactor (Fixed Component of O&M costs)	$60/kWe-y (from Module R2)	$85/kWe-y (from Module R2)

Capital and O&M Costs of the IC Fusion Plant. Over the 40+ years that inertial fusion energy has been pursued, there have been numerous cost estimates and cost models developed for fusion power plant concepts. The cost estimates for a pure fusion plant are the most mature, but it must be remembered that

they are based on an assumed regulatory structure that is not as costly and time-consuming as the regulatory system likely for fission power plants (or for FF hybrids). The LLNL costing tool is proprietary. It incorporates costs for the plant (structures and conventional power plant equipment) by an Architect & Engineering firm in 2012, laser costs based on NIF construction experience and vendor bids (e.g., laser diodes), and costs based on engineering estimates for other fusion-specific components.

LLNL's model includes algorithms for moving from direct capital cost to overnight capital cost, annual maintenance and staffing costs, depreciation, expenditures of capital funds over time, etc.

For the fusion system, the following are the top level cost areas:

1. Owners cost (project management and licensing)
2. Fusion Operations Building (includes the mechanical and electrical plant)
3. Plant Support Operations, Facilities, and Improvements (includes site works, electrical substations, administrative building, plant support building and process gas plant, and laser beam prep building)
4. Supervisory Control System
5. Fusion Engine (includes fusion chamber, vacuum chamber, chamber gas and fusion debris recovery system, primary coolant loop, secondary cooling loop, inert gas system, engine maintenance building, and remote maintenance equipment)
6. Fuel Injection, Tracking and Engagement System
7. Laser System (includes special equipment, beam path infrastructure, fusion operation building beam path infrastructure, and laser cooling system)
8. Procured Fuel Components
9. Fusion Fuel Operations (includes 50% of tritium plant and fuel handling building, spent hohlraum recycle and disposal, and on-site target manufacturing system)
10. Tritium Plant Equipment (includes 50% of tritium plant and fuel handling building, and tritium plant equipment)
11. Power Conversion Island including Structures (includes turbine generator equipment)

Costs are estimated one or two levels (code-of-accounts) below this top level, accumulated to this level, and then processed through a time of expenditure algorithm to obtain the capital costs. Operation and maintenance costs are built at a similar level of detail. References R9-2.9 and R9-2.10 describe the LLNL cost model and the influence of design parameters on its results.

The previously mentioned papers on ICF hybrids do not include capital or operating cost estimates. However, Anklam (Ref R9-2.9) provides some cost information for a pure fusion IC plant. Three plants are discussed with fusion powers of 400, 2200 and 2660 MW. The thermal power of these pure fusion plants would be about 20% higher. The 2200 MW fusion plant corresponds to 1 GWe of net power. The cost of these plants ranges from $4-6B. Assuming the middle of the cost estimate range ($5B) corresponds to the 1 GWe plant gives $5000/kWe. Operating costs are not shown, but assuming a fixed O&M cost of ~1% of the capital cost results in $50/kWe-y. This is slightly lower than the $60/kWe-yr used for the subcritical systems. These are the values used for the "Low Cost" entries in Table R9-2.2.

Similar to the assumption for a MCFR, the authors of this module assign a "High Cost" of twice the pure fusion model reference cost to reflect design uncertainty, cost uncertainty, and the economy of scale penalty of a (likely) smaller fusion engine appropriate for the FF hybrid application.

Table R9-2.2 What-it-Takes Cost Range for the IC Portion of a FF Hybrid System (2012$).

Item	Low Cost	High Cost
Inertial Fusion Portion of the FFH system (Specific Overnight Cost including contingency)	$5,000/kWe NOAK	$10,000/kWe NOAK
Inertial Fusion Portion of the FFH system (Fixed Component of O&M costs)	$50/kWe-y	$100/kWe-y

The possibility that the fusion system will not achieve its stated design performance in terms of net power output and capacity factor is not considered here, as successful development of the concept must be assumed to provide the basis for NOAK costs. However, successful development is a valid concern given the low technology readiness level of the concept and the significant technical issues remaining.

R9-2-7. DATA LIMITATIONS

No FFH system or continuously-operating fusion power plant has been constructed and operated to date; therefore, the cost assumptions presented here are largely estimates of costs based on paper studies for hypothetical systems. Further, the rigorous licensing regime for fission system, necessitated by the stored thermal energy in the fission products and the need to limit criticality excursions, will apply to FF hybrid systems, but will not apply to pure IC fusion systems. It is again noted that the costs for a dedicated FF hybrid design may be considerably different than those generated by costing the two sides of the plant separately.

The cost uncertainties for ICFRs are much higher than for fission reactors of any type. This is a result of the much lower technical maturity of controlled fusion vis-à-vis controlled fission. (Inertial fusion facilities have not yet reached the goal of plasma ignition that is a rough analogy to achieving criticality in the first fission reactors during the early Manhattan project of the early 1940s). Ref. R9-2.11 discusses the very considerable R&D needs for commercial fusion; this is one of a series of reports by that group, which usually focused on MC, but did consider IC to a lesser extent. Ref. R9-2.3, from the LLNL pure IC fusion project, discusses IC R&D needs. Ref. R9-2.12 discusses the R&D needs for FFH systems. Ref. R9-2.13 presents the views of skeptics of FFH technology.

R9-2-8. COST SUMMARIES

The specific costs of the two major components of an IC FFH system are summarized in the “What-it-Takes” Table R9-2.3. In order to calculate the specific costs for a complete FFH system, the user (fuel cycle analyst in this case) must define the mission, material balance, and energy balance for the whole system. (The formula for apportioning cost by power output is discussed in Section R9-2-6.)

The operation and capital costs of both the subcritical blanket (reactor) and pure fusion portions of the FFH are higher than those of most critical reactors, mostly because of the added costs of the fusion system and its integration with the fission portion, and also because of the higher complexity and integration requirements of the subcritical system. .

For the NOAK case, the subcritical reactor part of the capital cost range has been estimated at 2100 $/kWe, to $6600/kWe by adding 10% to both the ALMR (Module R2) low and high “What-it-Takes” specific costs. The low end of the ICFR range is the reference cost reported for a recent pure fusion design, using the LLNL proprietary cost model. The high-end value is twice the low-end value. The factor of two is meant to account for design uncertainty, cost estimate uncertainty, and economy of scale penalties associated with a smaller fusion engine in the case of a FF hybrid.

The “What-it-Takes” fixed component of the O&M costs was derived in a similar manner, with the ALMR (Module R2) providing the low end of the range, and with the ADS (Module R6) providing the high end of the range, for the subcritical reactor. For the inertial fusion portion of the hybrid, reference

R9-2.9, based on information for a recent pure fusion design, provides the low end of the range, with doubling for the high end of the range. Mode values were selected near the midpoints of these ranges.

Table R9-2.3. What-It-Takes Cost Summary Table for an IC FFH System (2012 AFC-CBD Update).

	Upside (Low Cost) 2012$	Nominal (Most Likely) 2012$	Downside (High Cost) 2012$
Capital Cost of the Subcritical Reactor	2,100 ($/kWe)	4,400 ($/kWe)	6,600 ($/kWe)
Capital Cost of the IC Fusion plant	5,000 ($/kWe)	8,000 ($/kWe)	10,000 ($/kWe)
O&M Cost of the Subcritical Reactor	60 ($/kWe-y)	100 ($/kWe-y)	230 ($/kWe-y)
O&M Cost of the IC Fusion plant	50 ($/kWe-y)	80 ($/kWe-y)	100 ($/kWe-y)

Table R9-2.4 shows the escalated values used for this 2017 AFC-CBD (1.088 escalation factor followed by rounding).

Table R9-2.4. What-It-Takes Cost Summary Table for an IC FFH System (2017$).

	Low Cost 2017$	High Cost 2017$	Mean 2017$	Mode 2017$
Capital Cost of the Subcritical Reactor	2,300 ($/kWe)	7,200 ($/kWe)	4,800 ($/kWe)	4,800 ($/kWe)
Capital Cost of the IC Fusion plant	5,400 ($/kWe)	10,900 ($/kWe)	8,300 ($/kWe)	8,700 ($/kWe)
O&M Cost of the Subcritical Reactor	65 ($/kWe-y)	250 ($/kWe-y)	141 ($/kWe-y)	109 ($/kWe-y)
O&M Cost of the IC Fusion plant	54 ($/kWe-y)	109 ($/kWe-y)	83 ($/kWe-y)	87 ($/kWe-y)

Figure R9-2.3 shows the resulting probability distributions and parameters for the costs in the Table above. The mean or “expected value” is also calculated.

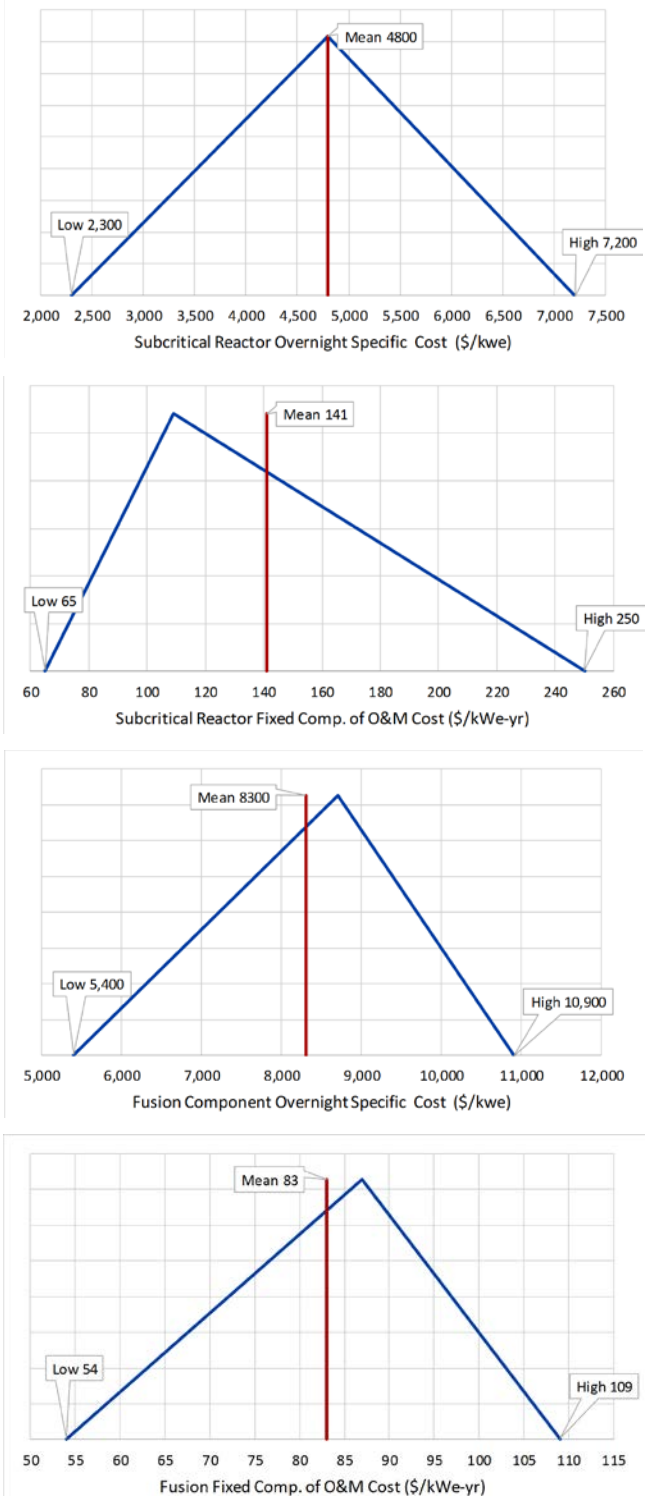

Figure R9-2.3. Probability Distributions for Hybrid Fission/Fusion System: Inertial Confinement Fusion

R9-2-9. SENSITIVITY ANALYSES

None available.

R9-2-10. REFERENCES

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R9-2.12 USDOE; Research *Needs for Fusion-Fission Hybrid Systems; Report of the Research Needs Workshop (ReNeW*); USDOE Workshop: Sept 30-Oct 2, 2009; Gaithersburg, MD

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