ELECTRIC POWER RESEARCH INSTITUTE

Program on Technology Innovation: Interim Progress on Two White Papers Supporting Advanced Reactor Commercialization

Expanding the Concept of Flexibility and Exploring the Historical Role of Public-Private Partnerships

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ABSTRACT

Public and private sector interest and investment in advanced nuclear reactor technologies is growing as utilities and other energy suppliers seek options for scalable, dispatchable, concentrated, and non-emitting energy sources. In keeping with EPRI's nuclear innovation mission and with its research portfolio increasingly focused on enhancing the flexibility, resilience, and integration of electricity and energy infrastructures, this report presents interim results from two separate studies. Both white papers included here are intended to inform and support the global advanced reactor research, development, and demonstration (RD&D) enterprise.

The first study attempts to refine and expand the concept of flexibility and identify potential metrics for evaluating advanced nuclear energy systems as future commercial options. To better reflect and capture the value of compelling attributes and expanded applications of advanced nuclear reactors, a broader concept of flexibility is proposed and defined, encompassing the following three key aspects: operational flexibility, deployment flexibility, and product flexibility.

The second study represents a historical assessment of government and industry roles to accelerate commercialization of advanced reactors. The assessment is based on the many parallel efforts to demonstrate and commercialize embryonic technologies during the 1950s and 1960s. Successful development and deployment of advanced reactor designs will likely require effective public-private partnerships similar to some of those early endeavors, which resulted in the commercialization of the current suite of nuclear technologies deployed around the world. This review captures lessons learned and key milestones in the historical RD&D paths that led to the incremental demonstration and deployment of light water reactors, heavy water reactors, and high-temperature gas reactors.

EPRI is sharing these interim work products in order to stimulate feedback and spur discussion among the advanced nuclear stakeholder community.

Keywords

Advanced nuclear technology
Advanced reactors
Flexibility
Public-private partnerships

DEFINITIONS

The following terms, acronyms and initialisms appearing in figures and text are defined as follows:

- AEA: Atomic Energy Act
- ARE: Aircraft Reactor Experiment
- BWR: Boiling Water Reactor
- CANDU: Canada Deuterium Uranium reactor
- GCR: Gas-Cooled Reactor
- GIF: Generation IV International Forum
- HTGR: High Temperature Gas-cooled Reactor
- HWR: Heavy Water Reactor
- LCOE: Levelized Cost of Electricity
- LEU: Low Enriched Uranium
- LWR: Light Water Reactor
- Magnox: *Magnesium Non-Oxidizing* (a Generation I gas-cooled reactor deployed in the United Kingdom named for the magnesium-aluminum alloy used for fuel cladding)
- MSR: Molten Salt Reactor
- NASA: National Aeronautics and Space Administration
- PWR: Pressurized Water Reactor
- PDRP: Power Demonstration Reactor Program
- RD&D: Research, Development and Demonstration
- TRL: Technology Readiness Level
- SMR: Small Modular Reactor

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1
INTRODUCTION

Public and private sector interest in advanced nuclear reactors is increasing as industry and governments look to the future and see the value of and need for energy options. EPRI launched a new strategic program in 2016 focused on advanced nuclear energy generation technologies. Many of these designs fall under the Generation IV designation and most incorporate primary coolants other than light water. EPRI's research, development and demonstration (RD&D) portfolio is increasingly focused on enhancing the flexibility, resilience and connectivity of today's power systems, while also drawing upon the lessons learned from past government-industry collaboration in the commercialization of nuclear technology.

This report presents interim results from two efforts to inform and support the global advanced reactor RD&D enterprise. The first attempts to refine and expand the concept of flexibility and identify of potential metrics for evaluating advanced nuclear energy systems as future commercial options. The second represents a preliminary assessment of government and industry roles to accelerate commercialization of advanced reactors based on the many parallel efforts to demonstrate and commercialize embryonic technologies during the 1950s and 60s.

With these interim products, EPRI seeks feedback from and dialogue with stakeholders. EPRI also seeks to inspire new lines of thinking to inform prioritization of and investment in relevant technology development while ensuring alignment of that development with societal and market needs.

However, in order to develop and deploy some of the advanced reactor designs that are currently under development, the lessons learned and shared successes from previous government-industry partnerships are worth revisiting and examining in the new context of today.

This historical analysis provides relevant data on selected milestones throughout the development of the light water reactor (LWR) and heavy water reactors (HWR) technologies, beginning with concept R&D performed by government, degrees of private sector contracted support, and ending with large-scale commercial implementations. In addition to the already operating water-based reactor, the historical assessment study also presents selected historical data for an additional set of power reactor technology concepts that were nurtured by the U.S. government, several with substantial private sector participation, but were not developed to the point of commercialization. This assessment focuses on the roles of government and industry during these projects.

2
FLEXIBILITY ATTRIBUTES AND METRICS

Many advanced reactor concepts are being developed globally for potential commercial deployment in the 2030 – 2040 timeframe. Unlike previous periods of reactor development, many of the developers in North America are small, entrepreneurial efforts with private backing, as opposed to the more traditional government driven efforts.

The commercial environment for new nuclear deployment is changing. Developed energy markets need to adapt large, aging infrastructures to maintain adequate energy and capacity, while addressing fragmented energy markets, disruptive competition from new sources of energy, and increasing regulation. Developing energy markets face the challenges and opportunities associated with “clean slates”, i.e., deploying new generation capacity and infrastructure where the need for such investment is clear. Furthermore, the established and proven technology, already demonstrated by large light water reactor (LWR) designs, is struggling to compete in many markets globally in terms of overnight costs, construction duration, levelized cost of electricity (LCOE), and other key economic factors.

EPRI’s research, development and demonstration (RD&D) portfolio is increasingly focused on enhancing the flexibility, resilience and connectivity of today’s power systems. In light of this holistic perspective, and consistent with continuous engagement with electric utilities and other key industry stakeholders, EPRI sees value in expanding the concept of flexibility as an important, if not essential, attribute of compelling, commercially relevant advanced reactor designs.

The Generation IV International Forum (GIF) originally established goals for reference Generation IV (GEN IV) concepts selected from among the many possible advanced reactor designs. These goals, aspirational in nature, have not been modified since the inception of GIF, but market outlook and external drivers have continued to evolve since that time.

As such, the GIF goals, while still relevant and desirable, do not adequately capture the range of features and attributes that may be required for a compelling case for commercial deployment alone. Given the current challenging market environments and the substantial capital costs associated with new nuclear construction, it is crucial that the most compelling and distinguishing features and attributes of advanced reactor technologies relative to current generation options be identified and valued appropriately. Table 2-1 presents the main features offered and the missions served by advanced reactors that potentially expand the competitive reach of advanced reactors beyond current technologies.

Note: The terms “GEN IV” and “advanced” are often used interchangeably when referring to reactor technologies beyond current Generation III/III+ designs, with most employing coolants other than water. However, the term GEN IV also carries the stricter, more limited definition established under GIF in 2002 for six reference designs and four goals. Therefore, the term “advanced reactor” will be used preferentially when discussing the more general set of non-LWR reactor technology options.

Table 2-1
Attributes and Missions for Gen IV Nuclear Energy Systems.

Features	Missions
• High temperature operation • Fast neutron spectrum (or thermal breeding for Th/U cycles) • Enhanced safety margins • Increased maneuverability	• Electricity generation • Process heat generation (incl. hydrogen and desalination applications) • Radioisotope production • Natural resource amplification • Waste management • Non-proliferation • Actinide Burning

Based on the attributes and diverse applications of GEN IV technologies, EPRI proposes expanding the use of the term "flexibility" to encompass three core capabilities:

- Operational Flexibility - the ability of an advanced nuclear system to be operated under a range of conditions through maneuverability (e.g., load following), fuel flexibility, compatibility with hybrid power systems and polygeneration, and remote/island operation.
- Deployment Flexibility - the ability of an advanced nuclear system to be licensed, financed, sited, and built under a wide range of external conditions through ease of scaling and siting, including the application of modular manufacturing and construction methods.
- Product Flexibility - the ability of an advanced nuclear system to fulfill more than one mission though product diversity (e.g., polygeneration).

The term "attribute", as used this document, relates to characteristics or inherent aspects relating to advanced nuclear technologies. A list of potential metrics is also presented for each flexibility sub-attribute that could be used to quantify, or assist with identifying, benefits of flexibility as part of the proposed design functions and capabilities of GEN IV reactors. A fundamental aspect of proposing metrics in this paper is to provide a basis to differentiate between options, in this case – advanced reactor concepts, to inform and aid the decision-making process.

2.1 Background

EPRI has developed decision analysis tools to inform RD&D planning and investment for alignment with industry needs, innovation opportunities, and resource limits. Through this activity the concept of flexibility was identified as an under-developed but potentially useful concept.

Expanding the concept of flexibility accommodates some attributes and missions that are not presently associated with nuclear generation. Some of these attributes are drawn from experience in other power generation technology sectors such as coal, natural gas, and solar thermal applications–where high temperature operations and combined-cycle power cycles are either deployed or are under active development. Likewise, products and missions other than those associated with electricity generation are drawn from apparent market trends and/or technology drivers.

The interest in flexibility arose from the identification of system characteristics that were not represented among the attributes used to evaluate advanced nuclear technologies during several EPRI-Vanderbilt Nuclear Fuel Cycle Assessment Workshops. While defining flexibility can be complex, the concept is intended to address a system's adaptability to uncertain operational, deployment, and market conditions. During workshops held in July and October of 2015, EPRI first introduced and then expanded the concept of flexibility as part of expert elicitations to prioritize of a broader set of attributes (Table 2-2 and Table 2-3, respectively) for future commercial advanced nuclear reactor concepts, using a software based decision analysis tool. Detailed descriptions of the elicitation approach, methods, software tool, and underlying framework are provided elsewhere.

Flexibility was defined as the ability to justify and/or adapt deployment and operation under challenging, changing, or uncertain external conditions and constraints, and to reliably fulfill one or more missions. Multiple levels of flexibility sub-attributes were developed for the July 2015 Workshop (Table 2-2); due to time constraints during the October 2015 workshop, the structure was simplified somewhat, to include only sub-attributes of flexibility ("Level 2" attributes, Table 2-3).

Table 2-2
Attribute Structure of the July 2015 EPRI Workshop Day 1 Elicitation.

Attribute (Level 1)	Sub-Attribute (Level 2)	Sub-Sub-Attribute (Level 3)
Waste Management	Minimizing Volume of Low-Level Waste	None
	Radiotoxicity of Repository Waste	None
Safety	Minimizing Worker Routine Dose	None
	Minimizing Factors that Drive Reactor Accident Risk	None
Economic Competitiveness	Minimizing Overnight Construction Cost	None
	Minimizing Operation and Maintenance Costs	None
	Minimizing Construction Duration	None
Flexibility	Operational Flexibility	Maneuverability of Reactor System
		Compatibility with Hybrid Technologies
		Ability to Use Different Fuels
	Deployment Flexibility	Scalability of System
		Compatibility with Siting
		Constructability
	Product Flexibility	Electricity
		Process Heat
		Hydrogen
		Radioisotopes
		Desalinated Water

Table 2-3
Attribute Structure of the October 2015 EPRI Workshop Day 2 Elicitation.

Attribute (Level 1)	Sub-Attribute (Level 2)
Resource Utilization	Natural Uranium and/or Thorium Required Per Unit Electricity Generated
	Fissile Content of Repository Waste Per Unit Electricity Generated
Safety	Worker Routine Dose
	Fuel, Operation and Maintenance Cost
	Construction Duration
Economic Competitiveness	Overnight Construction Cost
	Fuel, Operation, and Maintenance Cost
	Construction Duration
Waste Management	Volume of Low-Level Waste (LLW) Destined for Near-Surface Disposal Per Unit Electricity Generated
	Radiotoxicity of Repository Waste Per Unit Electricity Generated
Flexibility	Operational Flexibility
	Deployment Flexibility
	Product Flexibility

Results from the two workshops are briefly summarized in the following discussion. Table 2-3 also presents overall rankings obtained for prioritization of attributes via expert elicitation. Experts from academia, government, utilities, national laboratories, consultancy groups, and technology vendors constituted the surveyed group of participants. Economic competitiveness and safety consistently dominated priorities for advanced reactor concepts, with each receiving a weighted ranking of one-third to almost half (Table 2-4), while other attributes like flexibility, waste management and natural resource utilization comprised the remaining portions in roughly equal proportions.

Table 2-4
Overall Results of Attribute Ranking from July and October 2015 EPRI Workshops.

Attribute	Prioritizations of Attributes and Flexibility Sub-Attributes (%)	
	July 2015 Workshop	October 2015 Workshop
Resource Utilization	not included	10
Safety	45	34
Economic Competitiveness	33	33
Waste Management	10	11
Flexibility	12	12
Operational Flexibility	5	5
Deployment Flexibility	4	4
Product Flexibility	3	3

Note: Flexibility Sub-attributes are broken down under the Flexibility Attribute and sum to 12%.

Review of rankings and comments from elicitation participants from both workshops did not show meaningful discrimination among the three flexibility sub- categories when studying the collective group responses. There was one notable exception among the utility representatives during the October 2015 workshop when it was apparent that utilities favored operational flexibility over deployment and product flexibility; thus this could reflect the relevance of flexible operations to the current commercial environment. Three themes emerged from comments provided during the two workshops, with respect to the consideration of flexibility as an attribute for evaluating advanced reactor technology:

- *Utility and importance:* Flexibility is an important potential differentiator among advanced nuclear generation technology options.
- *Correlation and interdependence of some sub-attributes within flexibility*: Some aspects of flexibility are connected and may be interdependent. For example, within operational flexibility, diversified fuel use may enable maneuverability due to enhanced fuel or cladding temperature capabilities. Also, considering operational and product flexibility, the ability of an advanced reactor system to support multiple products may be correlated with that system's ability to operate as part of an integrated hybrid energy system.
- *Correlation between flexibility and some other attributes*: Some aspects of flexibility may be associated with economic competitiveness of an advanced reactor system. Accordingly, some participants recommended designating flexibility as a sub-attribute under economic competitiveness.

These comments concerning interdependence of flexibility with other high-level attributes, along with other feedback received, were taken into consideration in the refinement of EPRI's definition of flexibility. That effort to refine the concept is the focus of the Section 2 of this report.

2.2 Proposed Flexibility Attribute Structure, Descriptions, and Metrics

The set of flexibility sub-attributes presented (and shown in Table 2-5) is a result of a number of refinements based on input from the two 2015 workshops and feedback from subject matter experts [EPRI 2014a, EPRI 2016, Krahn 2014] (see above), as well as a review of work completed by the GEN IV International Forum (GIF) [GIF 2002, GIF 2014] and by the Department of Energy's Fuel Cycle Option Evaluation & Screening Effort (DOE E&S) [Wigeland 2014]. More importantly, Table 2-5 presents a proposed flexibility paradigm for advanced reactor technology assessments that will start to establish and clarify working terms of potential benefits of advanced reactor technologies.

Table 2-5
Proposed Flexibility Attribute Structure.

Sub-Attribute (Level 2)	Sub-Sub-Attribute (Level 3)
Operational Flexibility	Maneuverability
	Compatibility with Hybrid Systems and Polygeneration
	Diversified Fuel Use
	Island Mode Operation
Deployment Flexibility	Scalability
	Siting
	Constructability
Product Flexibility	Electricity
	Process Heat
	Radioisotopes

To further develop and define the flexibility sub-attributes, a literature review was conducted to describe the background and nature of these concepts to support the establishment of consistent definitions. Evidence from non-nuclear applications is introduced in these sections when the literature is not adequately described for the nuclear case studies alone. Furthermore, this document is ultimately intended to support future decision-making exercises. Therefore, this paper strives to suggest metrics that could be used to evaluate and distinguish technologies. These metrics are not quantitative in every instance, but the goal is to develop a measure that can at least be intuitively rated (on some qualitative scale). The literature review presented is also intended to summarize the reports pertaining to the concepts and guidelines, which relate to the developed attributes. The intent is to develop additional resources to be applied to the development of metrics of the flexibility attribute and its sub-attributes.

2.3 Operational Flexibility

The operational flexibility of an advanced reactor system is its ability to accommodate a wide range of external conditions during its operational lifetime. These conditions include physical demands and variations in the electrical grid, as well as energy market conditions. The four sub-attributes nested under operational flexibility are maneuverability, compatibility with hybrid energy systems (and polygeneration), diversified fuel use, and nuclear island mode operation.

2.3.1 Maneuverability

The maneuverability of an advanced reactor system is the ability to change its power level and corresponding electrical outputs in response to changing operational requirements and external conditions, including electrical load following. This concept has been applied by utilities and has been reviewed in the literature. Electricity is presently difficult to store in large quantities (either directly or indirectly); therefore, typically, it must be generated and supplied in real time based on the demand. Electricity demand can vary from moment to moment, hour by hour, day to day, and season to season [EIA 2013]. The need for power system maneuverability is driven by several trends:

1. The deployment of renewable energy systems (RES) has led to large temporal fluctuations in demand, which can be variable and unpredictable.
2. Operational concerns have increased due to frequency deviations caused by transient power imbalances due to market changes.
3. The emergence of a “smart grid” vision could be a driver for changes in power system operations [Ulbig 2015].

Maneuverability of a power supply may be required based on:

- Growth in variable generation (for example renewables such as solar photovoltaic and wind), which drives the need for flexibility based on its variable nature. This proves challenging to provide reliable power system operation.
- Environmental regulations, which can place constraints on conventional generation operations and lead to generator retirements.
- Impact to Independent System Operators (ISO)/ Regional Transmission Organizations (RTO) power markets and utility system operations through their commitment to flexible generation.
- Consumer adoption of energy efficient items such as electric vehicles, zero net energy buildings, energy efficient appliances which may lead to a more variable load, sharper peaks, or may provide extra flexibility [EPRI 2014b].

Power systems must be designed to ensure spatial and temporal balancing of generation and consumption (supply and demand) at all times. Maneuverability allows for a power system to maintain continuous service regardless of the need to meet rapid and large swings in demand. Maneuverability can be considered for:

- Short-term needs - balancing markets with a timeframe of up to one hour
- Mid-term needs - spot markets, up to days
- Long term needs - future contracts, seasonal variations [Ecofys 2014].

Maneuverability consists of two operational power modes: frequency control operations and predefined power maneuvers. Frequency control operations are used to control the frequency of power on the grid by operating within a set of predefined conditions directed by an ISO. This can be achieved by primary control, which is used to stabilize frequency transients, and secondary control, which is used to restore grid frequency on a scheduled value after grid frequency transients occur.

Possible Metrics for Maneuverability: To measure the ability of a design to achieve maneuverable operation, the firm Ecofys has considered the following attributes and associated metrics of "supply flexibility", which is essentially an alternate phrasing of maneuverability [Ecofys 2014]:

- Efficiency
- Reaction Time
- Investment Costs
- Variable Costs
- Installed Capacity
- Minimum Load
- Lifetime
- Maximum energy content/Maximum period of shifting
- Maturity of Technology
- Environmental Effects
- Economic Barriers
- Potential Role

Some of the Ecofys supply flexibility metrics are not pertinent to comparisons between advanced reactor systems; however, several of the metrics are potentially viable measures for the maneuverability sub-attribute. The most relevant options are reaction time (in operation and from cold start), the minimum load, and environmental effects/economic barriers (lumped together for the time for reasons described below):

- Reaction time is the most important characteristic and is defined as how quickly can an advanced reactor system can respond to changes in energy demand. This involves how quickly systems can come online from a cold start or how quickly they can ramp up or down to match the demand and is measured in percentage (of the total power level change required) per unit time.
- Minimum load is a measure of the lowest threshold of power that a system can operate, without severely disrupting long-term operations.
- Environmental effects and economic barriers are the environmental and economic burdens, respectively, introduced when a system shifts from baseload production to operations with continuously variable electricity outputs. Ramping (both up-and-down) operations can compromise efficiency in the short term, which in turn can compromise product yields per unit of time and per unit of fuel. In the longer term, frequent variations in power level will likely accelerate the rate of wear and tear on the unit [Ecofys 2014]. These tendencies impact both environmental and economic performance in a correlated fashion, since lower product outputs can result in larger environmental impacts on a product-normalized basis, and wear and tear could increase exposure rates due to maintenance and could potentially impact the safety of the system.

2.3.2 Compatibility with Hybrid Systems and Polygeneration

This sub-attribute describes the ability of an advanced nuclear system to be used in conjunction with a hybrid technology to produce one or more energy commodity outputs – co- or poly-generation [EPRI 2013. The combined system can operate in steady-state or transient modes depending on system needs and requirements [Antkowiak 2012, Bragg-Sitton 2016]. Electricity produced by advanced reactor systems cannot be directly stored using current technology at the required scale . Therefore, the supply of electricity must be adjusted constantly to match demand, as discussed in the previous section. Electricity needs to be available at both high-peak and off-peak demands. However, at off-peak times, there may be a surplus of energy, unless the ability exists to store the surplus energy. Two methods of storing energy are by thermal storage [Antiowak 2012, Bragg-Sitton 2016, Forsberg 2014] or by chemical storage [Antkowiak 2012, Bragg-Sitton 2016].

Thermal energy storage can utilize materials such as ceramic bricks or water when excess energy is available. This technique allows utilities to charge energy storage devices with excess generated electricity when demand is low. Later, when energy is required, it is extracted from the stored heat [Carnegie 2013]. Thermal energy storage systems mainly consist of three parts, the storage medium, heat transfer mechanism, and containment system. Thermal energy storage systems store the thermal energy in the form of heat (sensible, latent heat of fusion or vaporization) or reversible chemical reactions. An example is thermal energy storage (TES) blocks [Kuravi 2013], which has typically been used in conjunction with concentrated solar power generation sources. In this case, the TES block stores the excess heat generated during peak power generation. Later, when energy production is low, the stored thermal energy is discharged to the electricity production system. An example of possibly synergy with a nuclear system entails with the pairing of a molten salt reactor with a lithium chloride-based TES block unit [Kuravi 2013].

Another example is the synergy of advanced nuclear systems with a Nuclear Air-Brayton Combined Cycle (NACC) and Firebrick Resistance-Heated Energy Storage (FIRES) system (a high-temperature firebrick heated system). NACC enables the use of auxiliary natural gas or stored heat to further elevate compressed air temperatures after nuclear system heating for the production of additional peak electricity. The FIRES system enables electricity to be purchased, when the price of electricity is below that produced by natural gas, and stored as heat in a reservoir constructed to fire bricks. That heat can be later used as a replacement for heat from natural gas for peak electricity production when prices are higher [Forsberg 2014]. In order for advanced reactor systems to be paired with hybrid technologies, the energy production and energy storage technologies must be compatible [Kuravi 2013].

Advanced nuclear systems can also be combined with chemical production systems. For example, an advanced nuclear system can be used to produce electricity during peak hours, but then in off-peak hours it can provide necessary process heat to a chemical facility with products such as hydrogen [Antkowiak 2012]. Hydrogen is a means to store energy in large quantities, unlike electricity which cannot be stored. Electricity and hydrogen are readily interchangeable, since hydrogen can be produced from advanced reactor systems and hydrogen can undergo re-conversion into electricity via fuel cells, combustion turbines, and reciprocal engines [Verfondern 2007, Kuravi 2013].

Possible Metrics for Hybrid Compatibility: There are two intrinsic reactor characteristics that could enable or limit the synergy of reactors with hydrogen production: the reactor outlet temperature and the power rating. The rationale for including these metrics is fairly simple; outlet temperature and total power are necessary to ensure the quality and quantity of heat eligible for storage, respectively. In addition to these intrinsic characteristics of the reactor itself, another factor to consider is the compatibility of the reactor technology with a particular energy storage system. For example, in hydrogen production, conventional electrolysis can be accomplished by a process heat temperature of 80^0C, while thermochemical water splitting requires a process heat temperature of 850^0C [Verfondern 2007]. The former approach would be compatible with virtually any reactor technology, while the latter could only be achieved in very high-temperature systems.

2.3.3 Diversified Fuel Use

This attribute describes the ability of an advanced reactor system to implement a variety of fuel designs, fuel materials, and fuel systems. Changes in operational parameters or evolving mission or safety requirements can lead to undesirable or unacceptable impacts on fuel system components and materials. The ability to introduce new fuel materials, configurations, and designs may enable larger operational envelopes without sacrificing performance and safety margins. Advanced nuclear systems can be designed to allow a variety of fuel types and fuel cycles to be used, based on compatibility with the system design and energy needs [Sowder 2016].

Development of new fuels for light water reactors, including potentially radical departures from the current UO2 – zirconium fuel system for enhanced accident tolerant fuel (ATF), requires careful consideration of multiple characteristics that are linked to safety, performance, and ultimately economic viability. These include [Bragg-Sitton 2016]:

- Burnup limits/cycle length (while maintaining criticality and fuel performance)
- Operational parameters (power distribution, peaking factors, safety margins, etc.)
- Reactivity coefficients and control parameters (shutdown margin, rod worths)
- Handling, transportation, and storage (consideration of fuel isotopics, handling dose, and mechanical integrity)
- Compatibility with existing infrastructure (e.g., fabrication facilities, loading, in-core operations, post irradiation handling and storage, etc., necessary to maintain acceptable economics).

However, a new fuel candidate may not be able to meet the previous five requirements for all reactor types. Thus, a more generalized argument is made to account for discrepancies between fuel introductions in a variety of reactor types. These objectives are as follows [Bragg-Sitton 2016]:

- Maintain or improve upon the thermal, mechanical, and chemical properties observed for the current state-of-the-art fuel systems.
- Provide accident-tolerant improvements that increase coping time (or grace period) under severe accident scenarios.

- Offer the capability for power uprate and increased burnup to allow an economic case to be made for the adoption of the new fuel system.

While these characteristics do help to better understand the qualifications needed to introduce new fuel into a nuclear system, they are rather general and do not draw on empirical data. In the following section on metrics, a table analysis method is proposed as a way of considering how fuels have been used or postulate for use among various reactor technologies.

Possible Metrics for Diverse Fuel Type Compatibility: Given the broad definition for this sub-attribute, there is no single metric (or even a small set of metrics) that encompasses each of the concepts related to flexible fuel use. The issue is that “fuel type” is a frequently- but inconsistently-used term that can refer to different heavy metal elemental compositions (e.g., Th v. U), different chemical forms (e.g., carbide vs. oxide), and even completely different fuel paradigms (e.g., solid vs. liquid fuels). It is difficult to address each of these important considerations simultaneously, so instead, they are addressed individually.

The four suggested measures of fuel “type” compatibility are:

- Elemental compatibility (e.g., U, U/Pu, Th/U, Th/Pu);
- Fuel phase compatibility (e.g., solid fuel, pebble bed fuel, and liquid fuel);
- Solid fuel chemical form compatibility (e.g., oxide, carbide, nitride); and
- Solid fuel cladding compatibility (e.g., zircaloy, stainless steel, silicon carbide).

The proposed method for analyzing these four primary fields involves creation of four tables, one for each fuel diversity measure, with the table columns corresponding to reactor types (e.g., PWRs, BWRs, gas-cooled reactors, etc.), and the table rows representing specific examples of the four main components for fuel diversity analysis.

2.3.4 Island Mode Operation

This attribute describes a nuclear system’s ability to operate in isolation from local, regional or national electricity distribution networks either on a routine or exceptional basis. This operational mode is also called “Island mode operation” and it encompasses two modes: stand-alone generation for “mission” loads and micro-grid support and parallel connection to the grid for independent, reliable power in the event of a larger system power outage. Island mode operation may arise from a several motivations, including remote location of energy demand, poor reliability of electricity grid, and unacceptable interruptions in offsite power supplies.

Many existing plants could, in principle, operate in an island mode if provided with appropriate modifications. Powering down to 20-30 %, using electricity to cover house loads (especially coolant pumps) and diverting excess steam to condenser are within the design margins of existing technology. Interest exists in using island mode capability to better manage the existing fleet as a national grid security asset, by using nuclear reactors to reenergize a grid that has been severely compromised for extended periods such that other generation assets like natural gas and coal, may not be available due to fuel supply disruptions and renewables cannot manage due to inherent limitations.

Possible Metrics for Nuclear Island Operation: Though there are a number of potential metrics for nuclear island mode operation spanning all the sub-attributes, the selection process can be narrowed in scope to specifically address metrics for reactor types that would be considered for deployment of a nuclear island. GEN IV reactors (including SMRs) that are targeting a potential use in a nuclear island mode type application, are focused on the following metrics: small thermal power output, modular design, minimal onsite maintenance, simplified operations, and long core lives [Sowder 2016]. Modular design, onsite maintenance, and operations concerns can be evaluated as areas under the deployment flexibility attribute, where those topics have been discussed in more detail. Thermal power output and core life are both metrics that depend on the specific scenario factors driving the use of a nuclear island in the first place. If political or socio-economic factors dictate that a region requires a certain amount of power output, the core and entire nuclear system need to be sized or designed to meet those needs. This, in turn, has a direct effect on the amount of onsite maintenance required, as refueling could often be avoided completely for the required amount of operating time.

While the proposed metrics of thermal power output and core life do provide a reasonable starting point for analyzing the potential of nuclear island operation, as previously mentioned, this subject would most likely be very dependent on the characteristics of a specific application. Long-term use in the development of third world countries versus short-term electricity generation in an area affected by natural disasters/war would differ in design specifications/operational requirements.

2.4 Deployment Flexibility

Deployment Flexibility is the ability for an advanced nuclear system to be deployed under a wide range of external conditions. The Deployment Flexibility sub-attributes (third-level) include:

- Scalability of System,
- Compatibility with Siting, and
- Constructability.

All three sub-attributes are discussed in the next pages of this section.

2.4.1 Scalability

The "scalability" sub-attribute represents the ability of advanced nuclear systems to be sized or resized to increase energy output to meet changes in demand; that is, the ability of advanced nuclear systems to increase output capacity to accommodate such growth. The scalability of a system subject to growing demand can be critical for long term, successful operation and use [Bondi 2000]. Two aspects to system scalability for advanced nuclear systems are:

- The ability of a system to increase its capacity with system modifications (e.g., power uprates), or
- The ability to use modular components to expand the capacity of the system.
- The ability of a technology to be designed, constructed, licensed, and operated at the desired scale.

Power uprates can range from small changes such as more precisely measuring the feed water flow (allowing for less uncertainty in reactor operation requirements) to significant modifications to major balance-of-plant equipment, such as high-pressure turbines, condensate pumps and motors, generators and/or transformers. Current research and development on uprating has focused on three main areas: fuel technologies, Nuclear Steam Supply Systems, and Balance of Plant. The concept of "Power Ultra-Uprates", which look to increase power production up to 25-50%, have been postulated but not yet implemented [DOE 2006].

Proposed small modular reactors (SMRs) design concepts can be built independently or as modules in a larger complex, with capacity added incrementally as required by demand increases. Units can be built off-site and then transported to locations for use, potentially reducing the amount of construction time required to upsize the facility [NEA 2011]. The ability to limit the amount of capital invested and incrementally meet rising energy demands has been advanced as an additional justification for SMR implementation.

Possible Metrics for Scalability: To evaluate this sub-attribute, two aspects of scalability must be considered. One is the whether or not the advanced reactor system is modular in design. If increased capacity is necessary, it would be called into question if the system could be incrementally expanded to meet demand. The metric could be evaluated on a "yes/no" bases with regard to individual advanced reactor systems. Also, an evaluation should be made to the capability of components to be updated to allow an uprate to occur. This would require evaluation of the types of materials and technologies available to update components of the advanced reactor system to increase electricity production. The metric would also be a yes/no evaluation.

2.4.2 Siting

Siting flexibility is described as the compatibility of an advanced nuclear system with a variety of host locations, environments, and conditions. Several considerations must be made to determine the acceptability locating an advanced reactor system. There are four major examples of previous efforts to evaluate the attributes of siting flexibility:

- The Public Service Commission of the State of Wisconsin developed a list of attributes to aid in the siting of power plants (not limited to nuclear). The attributes were developed to represent concerns of various stakeholders, including utilities, independent power producers, regional planners, government agencies, various public interest groups, and the public. Six attribute categories (site requirements, community impacts, public health and safety concerns, environmental impacts, land use impacts, and economic impacts) were addressed [PSC 1999].
- The IAEA also developed a set of attributes to consider when determining a suitable site for a nuclear power plant. The IAEA's four attribute categories were health/safety/security, engineering/cost, socio-economic, and environmental [IAEA 2012a].
- EPRI developed a siting guide that includes site selection and evaluation criteria for new nuclear reactors. The guide takes the reader through a five-step process that addresses four groups of metrics: health and safety, ecological, socioeconomic, and engineering-cost [EPRI 2015].
- Oak Ridge National Laboratory (ORNL) has developed siting criteria for the evaluating existing coal plant sites, which could be adapted for small modular reactors (SMRs). The criteria included population density, ecology, flood risk, water access, earthquake risk, and the proximity of hazardous facilities (e.g., refineries) [ORNL 2013].

Another important aspect of siting flexibility particular to the nuclear industry is that site planning must also take into account emergency planning zone (EPZ) land requirements. The EPZ is the circular area encompassed by a 10-mile radius that surrounds the nuclear power plant, as codified in 10CFR50.47, “Emergency Plans” that states [NRC 1980]:

> Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

Possible Metrics for Siting: There are numerous aspects which could considered, but it is important to limit the metrics to a manageable set, as observed in the previous elicitations and work to prioritize high-level attributes (see Section 1.2). Aspects to consider in evaluation of advanced nuclear systems and siting might include:

- Ecological footprint
- EPZ requirements imparted by the current regulatory structure.
- Access to water – important for designs which require significant amounts
- Proximity population centers – must be sufficiently close to areas of electricity demand, but also where public acceptance and “consent-based” siting are surmountable
- Access to transmission lines

The evaluation of siting flexibility under deployment flexibility could consider ranges that relate to a qualitative scale (less favorable, moderately favorable, and more favorable). The evaluated advanced reactor system seen as having the least impact to the siting criteria or best benefit should be the most preferred.

2.4.3 Constructability

The constructability sub-attribute describes the relative ease with which advanced nuclear systems can be built. In this paper, constructability is defined as the ease with which a system can be built, based on its quality and completeness of design, and ease of implementation [Gambatese 2007]. Constructability is a concept used in construction management and has a variety of definitions. The concept of constructability has also been implemented in the construction of fossil fuel power plants, where it has been used for a variety of purposes such as providing reductions in project schedules. The underlying philosophy is that site layout should promote efficient construction, operations, and maintenance [O’Connor 1993]. Furthermore, design and procurement should consider the following objectives:

- Schedules should be informed by construction considerations
- Designs should be configured to enable efficient construction
- Design elements should be standardized
- Technical specifications should promote construction efficiency

- Detailed designs of modules and prefabricated components should be prepared to facilitate fabrication, transport, and installation.
- Project designs should promote accessibility of construction personnel, materials, and equipment [O'Connor 1993].

Utilizing specifications and designs that involve industry-standard products, equipment, and methods can improve the constructability of a project. Overly restrictive specifications can result in nonstandard requirements. Standardization of design can reduce unknowns that can affect the implementation of an advanced reactor system. Additionally, utilizing prefabricated structures can enhance the economic outlook for construction [O'Connor 1993]. For instance, the "1-3-8 Rule" is used as a casual rule-of-thumb in naval shipbuilding to indicate that on-site construction is eight times more costly than factory-setting construction, and three times more costly than staging area assembly [Seubert 1988].

A particularly important characteristic to examine when analyzing a large system's constructability is its degree of modularity, which MIT defines as "building complex products from smaller subsystems that can be designed independently yet function together as a whole" [MIT 2000]. An example of reactor modularization is the AP1000 design from Westinghouse [NYT 2011]. A key benefit to modularity is the ability to delay design decisions until more information is available without delaying the product development process [Gershenson 1999]. This is important in the context of advanced nuclear systems, as the field is dynamic and changes are constantly being implemented in modern designs.

With respect to modularization, it is recommended for design modules to be as large as possible so as to further simplify final assembly. Having many small, prefabricated elements can increase the number of trips required, which increases cost [O'Connor 1993]. However, the size of these modules must be reasonable so standard equipment can be utilized for transport and loading/unloading [O'Connor 1993]. Also, elements of exceptional size and dimensions require special care when being transported and may require additional site logistics and assembly work.

However, the use of modularization techniques should be implemented with careful consideration. An example of a potential pitfall of excessive modularization is the production of Boeing's 787 Dreamliner, which has been criticized for its overly heavy reliance on outsourced labor, which in turn has led to components that have failed to successfully function as an integrated system [Reuters 2009, HBR 2013]. Therefore, while modularization can benefit production, it must be implemented with due consideration to integration requirements.

A related topic is the percentage of off-site component fabrication. Quality assurance can be enhanced if fabrication of larger modules is done in a location suitable for manufacturing [Talabi 2012]. The factory environment allows for more precise standardization of materials, with higher quality welding and fabrication techniques [Westinghouse 2016]. However, just like in modularization, there needs to be careful consideration when outsourcing production, as having a variety of parts brought in from various sources can lead to issues with system uniformity and lack of proper integration.

Possible Metrics for Constructability: Developing a series of metrics to describe important characteristics of constructability is a challenge, as the nature of these metrics is somewhat qualitative. Advanced nuclear systems are so complex in nature that there are benefits to determining if the fundamental production logistics of such large-scale systems can be enhanced by appropriate attention to these new constructability characteristics. On the most basic level, a general system can be developed for the two key characteristics (modularity and off site fabrication), that asks the question, “Can it be done?” for which the answer would be either “yes”, “no”, or “possibly”.

However, work has been done (especially in the area of modularity) to obtain a more systematic understanding of the degree to which constructability can be achieved. Measuring a system’s degree of modularity begins by generating a component tree and process graphs to determine the various subsystems and how they are utilized. This leads to the creation of a modularity evaluation matrix that compares a variety of components and processes to determine how they are interrelated [Gershenson 1999].

2.5 Product Flexibility

Product Flexibility is defined as the ability of advanced nuclear systems to diversify their outgoing products. Detailed, third-level attributes are defined for Product Flexibility with respect to production of:

- Electricity,
- Process Heat, and
- Radioisotopes

2.5.1 Electricity Generation

The electricity sub-attribute represents a nuclear system’s ability to generate electricity, most commonly in the context of supplying electrical power to the grid. This section describes the product flexibility sub-attribute, so it is important to recall that flexible power operation topics, such as load-following, are covered in the maneuverability section under the operational flexibility sub-attribute. Other attributes in the EPRI Decision Framework consider the economic considerations of the design and implementation of advanced nuclear systems [EPRI 2014a, EPRI 2016]; thus, financial topics are not addressed here either. Instead, this section will focus specifically on baseload electricity generation as a product of a nuclear system.

Baseload power is the minimum electrical output needed to supply demand from the grid over the course of a day [Dutzik 2011]. While a number of nuclear power plants have developed the ability to vary power output based on demand such as in Germany and France [NEI 2012], historically, nuclear, hydro, and fossil fueled energy sources have continuously provided nearly their maximum output, with minimal variation. This means that during times of low energy demand (e.g., the middle of the night in a temperate climate), energy is ideally supplied by only baseload power plants.

Separating the flexible power operation from this particular discussion (as discussed above), there are several system characteristics that could serve as viable metrics for a nuclear system’s ability to produce baseload electricity. First, the capacity, or total rated power of the system, is the simplest way to examine a nuclear system’s ability to generate electricity. Total rated power is defined the amount of power a nuclear power plant is capable of producing, however, there are a number of variations of the term “capacity” that need to be defined for the sake of clarity. Working from the core of the reactor outwards towards the grid, the first available capacity metric is the thermal capacity (“MWt” or “MWth”) of the reactor, which takes into account the energy from heat associated with steam generation from fuel fissioning. After the attached steam turbine and generator convert the heat/steam to electrical energy, gross electrical capacity (MWe) is taken as a separate metric. The conversion between thermal and gross electrical capacity is directly proportional to the thermal efficiency (ηth), as governed by the equation: $MWe = MWth \times \eta_{th}$.

For most current reactors, thermal efficiency values usually range from around 33-37%; however, many advanced high-temperature reactors (AHTR) promise figures in the 40-50% range. Work done by Oak Ridge National Laboratory (ORNL) on advanced reactors such as molten salt reactors (MSR) and high temperature gas cooled reactors (HTGR) indicate that thermal efficiencies can reach into the range of 50% [Forsberg 2004]. More specifically, the AHTRs studied by ORNL use a molten salt coolant, gas turbine cycle that allows for 2400 MWt to be converted to 1300 MWe, a thermal efficiency of 54% [ORNL 2004]. Lastly, before power is sent to the grid, a certain amount of electrical energy is removed in order to operate a variety of components in the system (i.e. control electronics, pumps, etc.). The power that is sent to the grid after deducting the power needed to operate the reactor is known as the net electrical capacity of the power plant (also measured in MWe) [WNA 2016]. The final term worth mentioning is nameplate capacity, which represents the maximum rated output of a system under certain conditions designated by the manufacturer, and is the metric often advertised when reporting the total power of a nuclear system [EIA 2016].

However, nuclear systems do not necessarily run at peak capacity at all times; thus it is important to examine capacity factor as well. Capacity factor is a ratio (represented as a percentage) of a power generating system’s net power output over a period of time to its nameplate output (ideal case) over the same period of time. Average capacity factors for LWR nuclear systems in the U.S. has been above 90% in the last two years, making nuclear energy the energy source with the highest capacity factor in the United States [EIA 2014]. Canada, whose fleet consists of CANDU pressurized heavy water reactors (HWR), had a total capacity factor of 78% during the period of 2010-2015. In the United Kingdom, the fleet of fourteen operational advanced gas reactors achieved an average capacity factor of 73% in the period from 2010-2015.

Specific power is an important attribute to consider when examining the potential for electricity production. Specific power is defined as a ratio of power to a unit of mass (usually of fuel or heavy metal). When analyzed in the context of nuclear systems, specific power can be improved by increasing thermal efficiency and fissions per mole of fuel [WIN 2016]. It has been stated that two of the primary objectives when designing the core for an advanced reactor system are maximizing core power density and attainable fuel burnup, which will increase specific power of the system and thus the potential to produce electricity [Greenspan 2010]. It is important to keep in mind that power density is slightly different than specific power (power density is power per unit of volume), but could potentially also constitute a viable metric. Though data is rather limited, different types of advanced nuclear systems can vary in specific power, with models such as the liquid metal fast breeder reactor and molten salt reactor having higher specific powers than the presently-implemented pressurized or boiling water reactor technologies [WIN 2016].

Possible Metrics for Electricity Generation: The previous section described three viable metrics for assessing the potential electricity production for a nuclear system; they include a variety of capacity metrics, capacity factor, and specific power or power density. Selecting from this list of metrics may depend on the specific nuclear systems that are being evaluated or compared. For instance, capacity factor is a potentially useful parameter, but it may be unknowable, or speculative, for systems that have not yet been commercialized. For example, test reactors have reported values of capacity factor, but the discontinuous nature of test operations results in a capacity factor that may not be indicative of predicted commercial reactor performance. Specific power or power density can be predicted with high accuracy even for systems that have not yet been constructed, but use of this metric implies that evaluators place at least some importance on mass or volume minimization, respectively, which may or may not be a priority for decision makers. A combination of these metrics may be the most appropriate measure for base load electricity production performance.

2.5.2 Process Heat Production

Process heat as a sub-attribute under product flexibility represents the ability of an advanced nuclear system to produce heat for industrial use and applications. In energy-intensive industries, input energy requirements can represent a considerable fraction of the final production cost. A constant supply of heat at an appropriate temperature is essential to support reliable production operations. To achieve this, a reliable, consistent supply of energy is required to maintain processing requirements. Advanced nuclear systems can generate heat that could be useful for industrial processes if:

- The nuclear power plant and the industrial facility are located in sufficiently close proximity,
- The nuclear power plant can reliably supply the required amount of heat,
- The nuclear power plant can supply the heat the needed temperature, and
- The heat cost is competitive with alternative sources [IAEA 1998].

Industrial process heat for use can be divided into two temperature levels:

- Low-temperature heat which includes hot water and lower-quality steam for district heat, desalination, and other purposes.
- High-temperature process heat, which includes process steam for various industrial applications or high temperature heat for conversion processes [Kupitz 1985].

Low-temperature nuclear process heat applications have been demonstrated to be technically feasible [IAEA 2012b]. Low-temperature heating can be accomplished through the use of warm water (100-130^{0}C), hot water (160-180^{0}C), and low temperature steam (up to about 250^{0}C). High-temperature nuclear heat applications can be divided into two types of techniques: the intermediate transfer using steam and the use of direct heat transferred through a heat-exchanger; the latter is capable of slightly higher temperatures (e.g., well over 900^{0}C) [IAEA 2012b]. The transferred heat can originate from within the advanced nuclear system or from waste heat from cooling systems [Podest 1998]. Though waste heat is the simplest source of heat output from an advanced reactor system, its potential use is restricted because of the low output temperature. Potential uses for this type of heating include warming of greenhouses or fields for agricultural applications as well as warming of water in fish ponds. Though the amount of heat available from waste heat utilization is high, applications are limited to the surrounding area near the advanced nuclear system [Podest 1998].

High-temperature heat can be used to support a variety of applications such as

- Shale oil recovery [Forsberg 2013];
- Coal gasification: can be accomplished in three ways: by steam coal gasification, by hydro-gasification, and by coal liquefaction. For steam coal gasification, coal is converted into synthesis gas (syngas), when it reacts with a gasification agent at temperatures greater than 800^0C. The gasification agent is either steam (steam coal gasification) or hydrogen (hydrogasification) [IAEA 2012b];
- Production of synthetic fuels and other hydrocarbons [IAEA 2007];
- Oil refining [Antkowiak 2012];
- Hydrogen production: hydrogen can be produced by either electrolytical or thermochemical pathways, both of which can employ process heat [IAEA 2012b];
- Chemical production [IAEA 2012b],
- Desalination: this function can also be achieved by allocating electricity, in addition to process heat. High temperature versions of multi-stage flash (MSF) and multi-effect desalination (MED) plants use saturated steam in the range of 100^0C to a maximum of 140^0C, though some plants can operate at temperatures as low as 60^0C [IAEA 2012b];
- District heat: most commonly used in climates with relatively long and cold winters. District heating can be accomplished by hot water or steam at $80\text{-}150^0$C

Required temperatures for selected applications are presented in Table 2-6.

Table 2-6
Industrial Processes and Required Temperatures.

Process	Minimum Required Temperature
Thermochemical processes for production of hydrogen	900^0C
Desalination	$100\text{-}140^0$C
Coal Gasification	$>800^0$C
Shale Oil Recovery	370^0C
District Heating	$80\text{-}150^0$C

Possible Metrics for Process Heat Production: Metrics for process heat production in nuclear systems can be grouped into three main categories: temperature of heat, proximity of application site, and reliability of heat production:

- The varying temperatures of discharged heat are important in order to determine the proper industrial applications. Gas reactors and molten salt reactors have the ability to operate at higher temperatures (600-1000°C) than other reactor technologies. In contrast, the more commonly used water-cooled reactors operate at relatively low temperature outlets (generally around 300-350°C). As mentioned in the section above, industry applications such as coal gasification and thermochemical hydrogen production rely on higher temperatures, and thus temperature can be an important metric [IAEA 1998].

- The proximity of the application site to the nuclear system which will be providing process heat is important. Waste heat is often discharged at relatively temperatures, and thus is not easily transported [DOE 2008]. Thus, the site would greatly benefit from being within reasonable proximity to directly transfer heat from the nuclear system to the application, if waste heat is to be used. Considerations for direct heat transfer would mostly focus on energy losses in the medium of transfer (piping, heat exchangers, etc.). A related potential metric is the carrier fluid for heat, as thermodynamic properties vary based on the medium and phase used for transfer and heat loss characteristics could make a significant impact on heat production viability.
- The reliability of a nuclear system as a heat source plays an important role, especially when comparing to potentially alternate sources. Nuclear systems tend to run 24/7 for most of the time (except during refueling), and thus are considered relatively consistent sources of heat. A system's capacity factor could be a useful surrogate measure of reliability (see capacity factor in the Electricity Production sub-attribute below), and help quantify whether or not economies of scale could be achievable to maximize use of waste heat. Another sub-metric that falls under reliability is the extent of chemical/radiological contamination of the waste heat stream [DOE 2008]. Interactions between chemicals in the heat stream and materials used in piping/heat exchangers can cause fouling, which corrodes the materials and harms the effectiveness of the system [DOE 2008].

2.5.3 Radioisotope Production

This sub-attribute describes the ability of an advanced nuclear system to generate radioisotopes as products for use in a variety of applications, such as the industrial, research, medical, and agricultural fields. Though some reactors are dedicated to the production of radioisotopes, these radioactive nuclides can also be produced as by-products in nuclear systems that primarily generate electrical power (or process heat for that matter). It is noted that radioisotopes can also be produced in accelerators and similar machines; however, this section will focus on radioisotope production via nuclear reactors. Nuclear reactors are most commonly used for radioisotope production due to their large volumetric capacity, their ability to simultaneously irradiate several samples, their (relative) economy of production, and the possibility to produce a wide variety of radioisotopes [IAEA 2003]. Some of the more commonly used non-medical radioisotopes and their applications include americium-241 (used in fire detectors), californium-252 and nickel-63 (both used to detect explosives), iridium-192 (used to test the integrity of pipeline welds), and carbon-14 (used in biological research) [NRC 2000]. The medical uses range from the diagnoses of pernicious anemia using cobalt-57 to iodine-131 used to treat thyroid disorders [NRC 2000]. However, the most commonly used radioisotope, accounting for about 80% of all isotope-based nuclear medicine procedures in the world, is Technetium-99 (Tc-99), commonly referred to with its parent isotope, molybdenum-99 [WNA 2016e]. Tc-99 is used in a large array of medical applications and will be the most commonly referenced radioisotope in this section due to its critical importance and consistently high demand.

There are a number of factors to consider when determining whether it is more beneficial to produce radioisotopes in research or power reactors. The primary benefit of radioisotope production in power reactors is the ability to generate the isotopes as a by-product, where power generation is the primary product or function. Power reactors do, however, present a set of challenges because of their commitment to feeding the grid with electricity. The two major pathways to isotope product generation are from targets and from separating fission products from spent nuclear fuel. In the case of target-based production, utilities may be hesitant to make the series of minor structural changes associated with creating space for target materials to be inserted, which is compounded by concerns with material interference near the reactor core. Targets in commercial reactors might require additional casing than would normally be necessary in a research reactor to insure against any potential interactions [ORNL 1965, IAEA 2003]. Commonly used encapsulation materials include aluminum, zircaloy, and stainless steel. These materials need to have a low absorption cross sections, good thermal conductivity, and short-lived neutron activation products for safety purposes [IAEA 2003]. Also, power reactor licensing may need to be modified to include isotope production (as was done for tritium production at Watts Bar [Pace 2004]). This could lead to higher upfront costs in order to ensure operational reliability.

One of the more prominent arguments in favor of using research reactors is their potential for a much higher neutron flux relative to power reactors, due to using fuel with a higher enrichment than the maximum enrichment allowed for commercial reactor. While precise comparisons depend on the core position in which the target is placed, in general, larger quantities of target material have to be placed in power reactors in order to irradiate equivalent amounts of the product isotope. Furthermore, commercial reactors would need longer irradiation times to achieve adequate isotope production, which in turn would require synchronization with the cycling of fuel in the power reactors. The high flux and specialized product mission of research reactors eliminates the myriad of issues present in power reactors. To-date, isotope-producing research reactors have consisted of two major types: enriched uranium, light water moderated reactors (LWRs), and natural uranium, heavy water moderated reactors (HWRs). A point of distinction between these technologies is their physical adaptability to radioisotope production; that is to say, the system's ability to provide easy access for insertion of target materials. It has been argued that LWR's provide easier access to the core via the open design; however, HWR's usually have remote manipulators built in to maneuver fuel rods, and thus have a simple mechanism to insert target materials for radioisotope production. In fact, the world's leading producer of medical isotopes, the National Research Universal (NRU) reactor on the Chalk River site in Ontario, Canada has long used a HWR design to produce much needed medical radioisotopes [Guerin 2010].

A problem currently unfolding in the field of radioisotope production is the dependence on current (and aging) reactors to carry the weight of radioisotope production. An example of particular concern is in the medical isotope field where two-thirds of the worldwide supply of Mo-99 comes from only two sources: NRU at Chalk River and the High Flux Reactor (HFR) in Petten, Netherlands [Ruth 2009]. The most notable issue with dependence to this extent is the inevitable shut down of both sites as a result of natural aging and international protocols which dictate finite life times for such nuclear systems. While many look to currently existing nuclear systems such as the University of Missouri Research Reactor to massively scale up their radioisotope production to compensate for the impending shut down of current isotope producers [Ruth 2009]. Others are considering alternative paths, even paths that replace the use of high enriched uranium (HEU) with low enriched uranium (LEU) [TRIUMF & UBC 2008], which has been driven by modern concerns of safeguarding special nuclear material [NAS 2009].

Logistically speaking, a number of factors need to be considered to determine whether or not a nuclear system would be capable of radioisotope production. Though there are claims that many radioisotope production features would have to be built into a nuclear system ahead of time, still others argue that incumbent systems could be slightly modified to be suitable for isotope production [IAEA 2003]. Specific components needed would include extra space around the core for target insertion, additional encapsulation structures, and devices for remote insertion of target materials (though in the case of HWR's these are often built in regardless, due to the need for remote fuel rod insertion) [IAEA 2003].

Possible Metrics for Radioisotopes: As a starting point for comparing advanced nuclear systems with regards to their ability to produce radioisotopes, questions to ask include whether or not a design could be adapted to radioisotope production and whether the effects of the target material's presence would cause operational issues. In terms of the reactor's flux and time considerations, a promising metric could be irradiation efficiency. Irradiation efficiency is "the ratio of activity produced in the target to the activity calculated using the basic growth equation" [IAEA 2003]. While this metric provides useful data on the system's efficiency, the value of the metric is compounded when irradiation time is also considered (which can also be calculated through a simple equation). Irradiation efficiency should not be confused with other potential metrics to measure a nuclear system's efficiency, such ratios relating uranium fuel burnup to total radioisotope production (in the case of fission product method) [DeVolpi 2014].

2.6 Summary of Proposed Flexibility Attributes & Discussion

Taking into consideration the attributes and expanded applications of GEN IV technologies, it is proposed here that the term "flexibility" encompasses three important considerations: (1) Operational Flexibility, (2) Deployment Flexibility, and (3) Product Flexibility. This paper more clearly defines the concept of flexibility and its sub-attributes, which were explored in previous work; it also seeks to remain technology-neutral.

Table 2-7
Flexibility Sub-Attribute Structure and Proposed Metrics.

Sub-Attribute (Level 1)	Sub-Sub-Attribute (Level 2)	Proposed Metrics for Sub-Sub-Attributes (Level 3)
Operational Flexibility	Maneuverability	Operational efficiency
		Reaction time (in operation and from cold start)
	Compatibility with Hybrid Systems and Polygeneration	Energy storage efficiency ("round trip efficiency")
		Life cycle energy requirements efficiency
		Energy/power ratio
	Diversified Fuel Use	Elemental compatibility
		Fuel phase compatibility
		Solid fuel chemical form compatibility
		Solid fuel cladding
	Nuclear Island Operation	Thermal Output
		Modular Design
		Minimal onside Maintenance
		Simplified Operations
Deployment Flexibility	Scalability	Modularity of design infrastructure for potential power uprate
	Siting	Footprint (including options to shrink/minimize the EPZ)
		Level of water requirements
		Need to locate near populated areas proximity to transmission lines
	Constructability	Degree of modularity
		% of off-site fabrication
Product Flexibility	Electricity	Electrical/thermal/nameplate capacity
		Capacity factor
		Power density/specific power
	Process heat	Temperature of heat
		Proximity of application site
		Reliability of heat production
	Radioisotopes	Primary purpose of nuclear system
		Irradiation efficiency
		Irradiation time
		Proximity to chemical processing facility

A list of potential metrics is also presented for each flexibility sub-attribute that could be used to quantify, or at minimum assist with identifying, benefits of flexibility as part of the proposed design functions and capabilities of GEN IV reactors (as shown in Table 2-7). A fundamental aspect of employing metrics in this paper is to provide a basis to differentiate between options, in this case – advanced reactor concepts, to aid the decision-making process.

This flexibility study describes an expanded concept of flexibility and its evolving definition, as guided by input from industry experts. With this examination of flexibility and its potential utility, we seek to spur dialogue among key stakeholders and inspire new lines of thinking to inform prioritization of and investment in technology development while ensuring alignment of that development with societal and market needs. As a result of initial socialization and review of the concept, several observations and conclusions can be made.

The expanded concept of flexibility as developed and described here is not necessarily independent of other key advanced GEN IV reactor technology attributes. In particular, flexibility appears to be linked with economic competitiveness, and a common recommendation from several expert elicitations was to recast flexibility as a sub-criterion of economic competitiveness. The three proposed flexibility sub-attributes (operational, deployment, and product flexibility) also share a number of potential interdependencies. Such interdependence must be considered during application of decision- analysis methods.

Flexibility alone is not an end in itself; however as discussed herein, it represents a set of options associated with deployable technologies as well as potential technology RD&D platforms and, pathways possess it to a greater or lesser degree. Over-emphasizing technology or system flexibility, however, without regard to other important attributes, could undermine efforts for commercial adoption.

EPRI is continuing to pursue development of an expanded concept of flexibility as an important and useful part of articulating a clearer, more compelling business case for advanced GEN IV reactor systems.

2.7 References

[Antkowiak 2012] Antkowiak, M., M. Ruth, R. Boardman, S. Bragg-Sitton, R. Cherry, L. Shunn. Summary Report of the INL-JISEA Workshop on Nuclear Hybrid Energy Systems. Technical Report NREL/TP-6A50-55650. July 2012.

[Archer 2007]. Department of Civil and Environmental Engineering, Stanford University. Supplying Baseload Power and Reducing Transmission Requirements by Interconnecting Wind Farms. November 2007.

[Azevedo 2011] University of São Paulo, Department of Metallurgical and Materials Engineering. Selection of fuel cladding material for nuclear fission reactors. December 2011.

[Blake 2006]. E. Michael Blake. U.S. Capacity Factors: Leveled Off at Last. May 2006.

[Bondi 2000] Bondi, Andre. Characteristics of Scalability and Their Impact on Performance. The Second International Workshop of Software and Performance. September 17-20, 2000.

[Bragg-Sitton 2016] S. Bragg-Sitton, M. Todosow, M. Montgomery, C. Stanek, R. Montgomery, W. Carmack. “Metrics for the Technical Performance Evaluation of Light Water Reactor Accident-Tolerant Fuel”. Special American Nuclear Society publication, February 2016.

[Brinton 2015] Brinton, S. (2015). The Advanced Nuclear Industry, Third Way.

[Buongiorno 2010]. MIT Center for Advanced Nuclear Energy Systems. PWR Description. 2010.

[Bush 1991] Bush, R. "Recovery of Platinum Group Metals from High Level Radioactive Waste: Possibilities of Separation and Use Re-Evaluated", Platinum Metals Review, Vol. 35, Issue 4, pp. 202-208, 1991

[Carnegie 2013] Carnegie, Rachel, Douglas Cotham, David Nderitu, and Paul V. Preckel. Utility Scale Energy Storage Systems: Benefits, Applications and Technologies. State Utility Forecasting Group, Purdue University. June 2013.

[Cobb 1976] Cobb, H. "Mechanical Tests of Full-Size Fort St. Vrain Fuel Blocks", General Atomics Report GA-A13917, 1976

[Davis 2013] American Nuclear Society. The Hook-Ons. April 2013.

[Denholm 2004] Denholm, Paul L. Environmental and Policy Analysis of Renewable Energy Enabling Technologies. Fusion Technology Institute, University of Wisconsin. December 2004.

[DeVolpi 2014] Nuclear Applications Company. Efficient and Timely Production of Valuable Radioisotopes (in Molten-Salt Reactor). 2014.

[Diesendorf 2015] EnergyScience Coalition. Do We Need Base-Load Power Stations? December 2015.

[DOE 2006] U.S. Department of Energy. Advanced Power Ultra-Uprates of Existing Plants. Report Number: STD-TFNE-06-10. March 31, 2006.

[DOE 2008] U.S. Department of Energy, Industrial Technologies Program. Waste Heat Recovery—Technology and Opportunities in the U.S. March 2008.

[Dutzik 2011] Frontier Group. What is Baseload Power? March 2011.

[Ecofys 2014] Papaefthymiou, Georgios, Katharina Grave, and Ken Dragoon. Flexibility Options in Electricity Systems. Ecofys Germany GmbH. March 10, 2014. Project Number PWDE14426.

[EIA 2013] U.S. Energy Information Administration. Electricity Demand Patters Matter for Valuing Electricity Supply Resources. Published August 27, 2013. http://www.eia.gov/todayinenergy/detail.cfm?id=12711

[EIA 2014]. U.S. Energy Information Administration (EIA). Monthly generator capacity factor data now available by fuel and technology. January 2014.

[EIA 2016a]. U.S. Energy Information Administration (EIA). U.S. Nuclear Operating Plant Basic Information. June 2016.

[EIA 2016b] U.S. Energy Information Administration (EIA). (2016) "Glossary" Accessed July 6, 2016 from https://www.eia.gov/tools/glossary/index.cfm?id=G#gen_nameplate

[EIA 2016c] U.S. Energy Information Administration (EIA). (2016, Last Revised December 3, 2015). "How much electricity does a nuclear power plant generate?" Accessed July 5, 2016 from https://www.eia.gov/tools/faqs/faq.cfm?id=104&t=3

[EIA 2016d] U.S. Energy Information Administration (EIA). (2016, Last Revised February 23, 2016). "What is the difference between electricity generation capacity and electricity generation?" Accessed July 6, 2016 from https://www.eia.gov/tools/faqs/faq.cfm?id=101&t=3

[EPRI 2013] EPRI, "Polygeneration: An Opportunity for Diversification and New Revenue". Palo Alto, CA: 2013. 3002002215

[EPRI 2014a] EPRI, "Program on Technology Innovation: Summary of 2014 EPRI Nuclear Fuel Cycle Assessment Workshop – Day 1: Trial Implementation of EPRI Decision Framework Vanderbilt University, Nashville, Tennessee, July 22–23, 2014", EPRI Report 3002002784, 2014

[EPRI 2014b] EPRI, "Metrics for Quantifying Flexibility in Power System Planning", EPRI Report Number 3002004243, 2014

[EPRI 2015] EPRI, "Advanced Nuclear Technology: Site Selection and Evaluation Criteria for New Nuclear Power Generation Facilities (Siting Guide)", EPRI Report Number 3002005435, 2015

[EPRI 2016] EPRI, "Advanced Nuclear Technology: Emergency Planning Zone Size Evaluation for Small Modular Reactors", EPRI Report Number 3002008037, 2016

[Forsberg 2004]. Oak Ridge National Laboratory (ORNL). The Advanced High Temperature Reactor: High-Temperature Fuel, Molten Salt Coolant, and Liquid-Metal-Reactor Plant. November 2004.

[Forsberg 2013] Forsberg, Charles. Hybrid Systems to Address Seasonal Mismatches Between Electricity Production and Demand in Nuclear Renewable Electric Grids. Energy Policy. 62, 333-341. 2013.

[Forsberg 2014] Forsberg, C., L. Hu, P. Peterson, and K. Sridharan. Fluoride-Salt-Cooled High-Temperature (FHR) for Power and Process Heat, Final Project Report. MIT-ANP-TR-157

[Gambatese 2007] Gambatese, J., J. Pocock, and P. Dunston, Constructability Concepts and Practice. 2007, American Society of Civil Engineers (ASCE).

[Gershenson 1999] J. Gershenson, G. Prasad, S. Allamneni. Utah State University Department of Mechanical and Aerospace Engineering. Modular Product Design: A Life-Cycle View. 1999.

[GIF 2002] Generation IV International Forum (GIF) (2002). A Technology Roadmap for Generation IV Nuclear Energy Systems. GIF-002-00

[GIF 2014] Generation IV International Forum (GIF) (2014). Technology Roadmap Update for Generation IV Nuclear Energy Systems, OECD Nuclear Energy Agency.

[Greenspan 2010]. University of California, Berkeley. Fundamentals of Nuclear Reactor Core Design. June 2010.

[Guerin 2010] Society of Nuclear Medicine and Molecular Imaging. Cyclotron Production of Tc-99: An Approach to the Medical Isotope Crisis. April 2010.

[HBR 2013] J. Allworth. Harvard Business Review. The 787's Problems Run Deeper Than Outsourcing. January 2013.

[IAEA 1998] IAEA. Nuclear heat applications: Design aspects and operating experience: Proceedings of four technical meetings held between December 1995 and April 1998. IAEA-TECDOC-1056.

[IAEA 1999] International Atomic Energy Agency. Production Technologies for Molybdenum-99 and Technetium-99. IAEA-TECDOC-1065. February 1999.

[IAEA 2003] IAEA, Manual for reactor produced radioisotopes. IAEA-TECDOC-1340. January 2003.

[IAEA 2007] IAEA, Non-Electric Application of Nuclear Power: Seawater Desalination, Hydrogen Production and other Industrial Application. Proceedings of an International Conference, Oaria, Japan, 16-19 April 2007. IAEA-CN-152.

[IAEA 2011] Power Uprate in Nuclear Power Plants: Guidelines and Experience Prepared within the Framework of the Technical Working Group on Life Management of Nuclear Power Plants.

[IAEA 2012a] International Atomic Energy Agency. Managing Siting Activities for Nuclear Power Plants. IAEA Nuclear Energy Series No. NG-T-3.7. 2012.

[IAEA 2012b] IAEA, Advances in Nuclear Power Process Heat Application. IAEA-TECDOC-1982.

[IAEA 2016] International Atomic Energy Agency. (2016, Last Revised July 7, 2016). "Power Reactor Information System (PRIS)", Accessed July 8, 2016 from https://www.iaea.org/PRIS/CountryStatistics/CountryDetails.aspx?current=GB

[Sowder 2016] A. Sowder, B. Burkhardt, S. Krahn, N. Irvin. Expanding the Concept of Flexibility for Evaluating Advanced Nuclear Energy Systems as Future Commercial Options. April 2016.

[INEEL 2004] Wei, Thomas Y.C. and Kevan D. Weaver. GEN IV Nuclear Energy Systems, Initial Requirements for Gas-Cooled Fast Reactor System Design, Performance, and Safety Analysis Models. Idaho National Engineering and Environmental Laboratory. INEEL/EXT-04-002242. August 2004.

[Kupitz 1985] Kupitz, Juergen and Milan Podest. Nuclear Heat Applications: World Overview. IAEA Bulletin, Vol. 26, No. 4, Pages 18-21.

[Kuravi 2013] Kuravi, Sarada, Jamie Trahan. D. Yogi Goswami, Muhammad M. Rahman, Elisa K. Stefanakos. Thermal Energy Storage Technologies and Systems for Concentrating Solar Power Plants. Process in Energy and Combustion Science. Volume 39, Issue 4, August 2013, Pages 285-319.

[Krahn 2013] S. Krahn, A. Croff, T. Ault, R. Wymer. Why Reconsider Thorium? 2013.

[Krahn 2014] Krahn, S., T. Ault, A. R. Gardiner, A. Croff, J. Clarke, A. Machiels and A. Sowder (2014). A Decision Analysis Tool to Support Planning and Decision-Making for Sustainable, Deployment-Oriented Research, Development, and Demonstration (RD&D) of Advanced Nuclear Energy Technologies. 2014 International Congress on the Advances of Nuclear Power Plants (ICAPP 2014). Charlotte, NC, USA: pp 711-719.

[Kwon 2015]. U.S. Energy Information Administration (EIA). Electric generator capacity factors vary widely across the world. September 2015.

[Lam 2007] Lam P., Chan A., Wong F. K. and Wong F. W. (2007). Constructability Rankings of Construction Systems Based on the Analytical Hierarchy Process. Journal of Architectural Engineering, 13(1): 36-43.

[Li 2014] INET, Tsinghua University. HTR Progress in China. April 2014.

[Low 2001] Sui Pheng Low (2001),"Quantifying the relationships between buildability, structural quality and productivity in construction", Structural Survey, Vol. 19 Iss 2 pp. 106 - 112 http://dx.doi.org/10.1108/EUM0000000005627

[Meyer 2006] Meyer, M., Fielding, R., and Gan, J. "Fuel Development for Gas Reactors", Submission to Journal of Nuclear Materials (INL/JOU-06-11592), 2006

[Mills 2015] Mills, Andrew, and Joachim Seel. Flexibility Inventory for Western Resource Planners. LBNL-1003750. October 2015.

[MIT 2000] C. Baldiwn, K. Clark. Massachusetts Institute of Technology. Design Rules, Volume 1: The Power of Modularity. 2000.

[NAS 2009] National Academies of Sciences, Engineering, and Medicine. Medical Isotope Production without Highly Enriched Uranium. 2009.

[NEI 2009] Nuclear Engineering International. Hyperion Launches U2N3-fueled, Pb-Bi Cooled Fast Reactor. November 2009.

[NEI 2012] Nuclear Engineering International (NEI). (2012, Last Revised June 1, 2012). "Load-Following Capabilities of NPPs", Accessed July 8, 2016 from http://www.neimagazine.com/features/featureload-following-capabilities-of-npps/

[NEI 2015] Nuclear Energy Institute (NEI) (2015). Hearing on The Future of Nuclear Energy December 11, 2014. U.S. House of Representatives. Committee on Science, Space and Technology Subcommittee on Energy. 113th Congress, 2nd Session. Washington, D.C.: GPO: 2015. (Statement of D.S. Lipman, Nuclear Energy Institute).

[NEI 2016a] Nuclear Energy Institute. Small Reactor Designs. 2016.

[NEI 2016b] Nuclear Energy Institute (NEI). (2016). "Electricity Supply" Accessed July 6, 2016 from http://www.nei.org/Why-Nuclear-Energy/Reliable-Affordable-Energy/Electricity-Supply

[NIT 2006] National Institute of Technology, Department of Physics (India). Production and Application of Radioisotopes. June 2006.

[NNL 2012a] National Nuclear Laboratory. Review of Metrics Relevant to Reactor Systems. NNL (11) 11491, Issue 3. January 2012.

[NNL 2012a] National Nuclear Laboratory. Review of Metrics Relevant to Reactor Systems. NNL (11) 11491, Issue 5. January 2012.

[NRC 1980] US Nuclear Regulatory Commission (NRC) (1980 (as amended)). 10CFR50.47 Emergency Plans.

[NRC 2000] U.S. Nuclear Regulatory Commission. The Regulation and Use of Radioisotopes in Today's World. NUREG/BR-0217. April 2000.

[NYT 2011] H. Northey. The New York Times. U.S. Nuclear Regulators, Westinghouse Spar Over AP1000 Safety Review. May 2011.

[O'Connor 1993] O'Connor, James and W.T. Norwich. Fossil Power Plant Constructability: Applications of CII Concepts. Journal of Energy Engineering, Volume 119, No. 1, April 1993.

[Oehme 1970] Oehme, H. and Schoening, J. "Design, Features, and Engineering Status of the THTR 300 MWe Prototype Power Station", Report CONF-700401, 1970

[ORNL 1965] Oak Ridge National Laboratory. Radioisotope Production in Power Reactors. ORNL-3792. May 1965.

[ORNL 2004]. Oak Ridge National Laboratory (ORNL). "Status of Preconceptual Design of the Advanced High-Temperature Reactor (AHTR). May 2004.

[ORNL 2013] Oak Ridge National Laboratory. Evaluation of Suitability of Selected Set Coal Plant Sites for Repowering with Small Modular Reactors. ORNL/TM-2013/109. March 2013.

[Pace 2004] Pace, P. "Watts Bar Nuclear Plant (WBN) Unit 1 - Tritium Production Program - Program Enhancements", License Amendment Request by Tennessee Valley Authority to the US Nuclear Regulatory Commission, 2004

[PNNL 2009] Pacific Northwest National Laboratory. Medical Isotope Production and Use. PNNL-SA-65456. March 2009.

[Podest 1998] Podest, Milan. Reactors for Low-Temperature Nuclear Heat Supply. Nuclear Engineering and Design. Volume 109, Issues 1-2, September-October 1998, Pages 115-121.

[Poullikkas 2013] Poullikkas, A. "A Comparative Overview of Large-Scale Battery Systems for Electricity Storage", Renewable and Sustainable Energy Reviews, Vol. 27, pp. 778-788, 2013

[PSC 1999] Public Service Commission, State of Wisconsin. Common Power Plant Siting Criteria. Published September 1999.
http://psc.wi.gov/thelibrary/publications/electric/electric05.pdf

[Raj 2000] Department of Atomic Energy, Government of India. Fuel and Materials Development for Prototype Fast Breeder Reactor. April 2000.

[Reuters 2009] K. Peterson. Reuters News Agency. Boeing 787 delays cast hard light on outsourcing. September 2009.

[Seubert 1988] T. Seubert. "Modular Shipbuilding and Its Relevance to Construction of Nuclear Power Plants." Master's thesis: Massachusetts Institute of Technology. 1988

[Talabi 2012] Talabi, S. and P. Fishchbeck. Advancing Risk Management in Nuclear Power Plant EPC Projects: An Empirical Evaluation of Risk Management Practices on Steam Generator

Replacement Projects. Proceedings of 2012 Seventh World Congress on Engineering Asset Management.

[TRIUMF 2008] Canada's National Laboratory for Particle and Nuclear Physics. World View of Radioisotope Production. August 2008.

[TRIUMF & UBC 2008] Canada's National Laboratory for Particle and Nuclear Physics and the University of British Columbia. Report of the Task Force on Alternatives for Medical Isotope Production. 2008.

[Ulbig 2015] Ulbig, Andreas and Göran Andersson. Analyzing Operational Flexibility of Electric Power Systems, International Journal of Electrical Power & Energy Systems Volume 72, November 2015, Pages 155–164.

[Verfonden 2007] Verfonden, Karl (Editor). Nuclear Energy for Hydrogen Production. Schriften des Forschungszentrums Jülich Reihe Energietechnik (Energy Technology). Volume 58. 2007.

[Weaver 2016] TerraPower at DOE-NRC Second Workshop on Advanced Non-LWR Reactors. Fuel Development, Testing, and Schedule for the Traveling Wave Reactor. June 2016.

[Westinghouse 2016a] Westinghouse Electric Corporation. AP1000 Nuclear Power Plant Construction. 2016.

[Westinghouse 2016b] Westinghouse Electric Corporation. Small Modular Reactor. 2016.

[Wigeland 2014] Wigeland, R., T. Taiwo, H. Ludewig, M. Todosow, W. Halsey, J. Gehin, R. Jubin, J. Buelt, S. Stockinger, K. Jenni and B. Oakley (2014). Nuclear Fuel Cycle Evaluation and Screening – Final Report. INL/EXT-14-31465

[WIN 2016] Whatisnuclear.com (WIN). (2016). "Energy Density Calculations of Nuclear Fuel". Accessed July 5, 2016 from https://whatisnuclear.com/physics/energy_density_of_nuclear.html

[WNN 2015] World Nuclear News. Russia Connects BN-800 Fast Reactor to Grid. December 2015.

[WNA 2016a] World Nuclear Association (WNA). (2016, Last Revised December 2012). "Energy for the World – Why Uranium?" Accessed July 5, 2016 from http://www.world-nuclear.org/information-library/nuclear-fuel-cycle/introduction/energy-for-the-world-why-uranium.aspx

[WNA 2016b] World Nuclear Association (WNA). (2016, Last Revised April 2016). "Nuclear Power Reactors" Accessed July 6, 2016 from http://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/nuclear-power-reactors.aspx

[WNA 2016c] World Nuclear Association (WNA). (2016, Last Revised September 2015). "Supply of Uranium" Accessed nJuly 5, 2016 from http://www.world-nuclear.org/information-library/nuclear-fuel-cycle/uranium-resources/supply-of-uranium.aspx

[WNA 2016d] World Nuclear Association. Nuclear Process Heat for Industry. February 2016.

[WNA 2016e] World Nuclear Association. Radioisotopes in Medicine. June 2016.

[WNA 2016f] World Nuclear Association. Fast Neutron Reactors. July 2016.

[Yasuda 2013] Yasuda, Yoh. "Flexibility Chart: Evaluation on Diversity of Flexibility in Various Areas." Wind Integration Workshop. London. October 2013

3
HISTORICAL ASSESSMENT OF GOVERNMENT-INDUSTRY ROLES IN THE NUCLEAR POWER COMMERCIALIZATION

This section describes a high level evaluation of the progression of governmental initiatives, industry responses, and the associated evolution of interfaces that eventually led to the successful commercial deployment of the following four technologies:

- Pressurized Light Water Reactor (PWR) in the U.S.
- Boiling Light Water Reactor (BWR) in the U.S.
- Pressurized Heavy Water Reactor (HWR - CANDU) in Canada
- Gas Cooled, Graphite Moderated Reactor (GCR - MAGNOX) in the U.K.

The historical record is examined to identify and describe the basic elements of successful path to commercialization, including:

- Basic concept research and feasibility assessment,
- Proof of concept through test reactor research,
- Construction and operation of small demonstration reactors,
- Construction and operation of large demonstration reactors, and
- Construction and operation of initial commercial reactors.

In each of these steps, this study identifies the methods used by governments to promote technology development and share project risk through such means as:

- Formulation of public policy promoting nuclear energy,
- Sharing of relevant technical data developed by the government,
- Financial sponsorship and cost-sharing,
- Collaboration on technology development,
- Construction of multi-purpose test reactors and associated facilities, and
- Provision of material resources.

The study further identifies the industry response to these government prompts through such means as:

- Private enterprise proposal of nuclear power technology projects,
- Formation of private enterprise partnerships and consortia to manage and absorb business and financial risk,
- Collaboration with government on identification of R&D needs,
- Engineering, construction, and operation of reactor test and demonstration facilities, and

- Increased acceptance of business risk with maturation of technology, leading to full scale operating facilities financed by private enterprise.

In addition to the focus on PWR, BWR, HWR-CANDU, and GCR-MAGNOX, this section provides a brief historical discussion of the roles of the U.S. government and private industry in early development of other reactor technology concepts including:

- Liquid Sodium-Cooled Reactor – Fast Spectrum
- Liquid Sodium-Cooled Reactor – Thermal Spectrum
- High-Temperature Gas Reactor (HTGR)
- Homogeneous Reactor and Molten Salt Reactor (MSR), and
- Pressurized Heavy Water Reactor (HWR)

Commercialization of safe and reliable reactor technology has historically benefitted from arigorous, sequential, and incremental progression toward technical maturity. The Technology Readiness Level (TRL) step-wise progression (see Figure 3-1, [DOE 2016]) of technology maturation is adapted from that utilized by the National Aeronautics and Space Agency (NASA) for high-stakes, high-rewards research, development, and deployment--and has been broadly accepted in high-hazard, complex applications in the aerospace and chemical industries in addition to the nuclear industry. Such high-stakes, high-rewards RD&D is arguably difficult for private industry to fully shoulder due to the considerable time and resources necessary to successfully advance novel and complex technology. The analysis in this section supplements the DOE conceptual treatment with concrete examples of the facilities that have made up each step of this sequential progression of technology maturity. It describes U.S., Canada, and U.K. historical cases where government and private industry partnered to bring about major technological advancement in generation of electrical power—which can inform the structure of future collaborations.

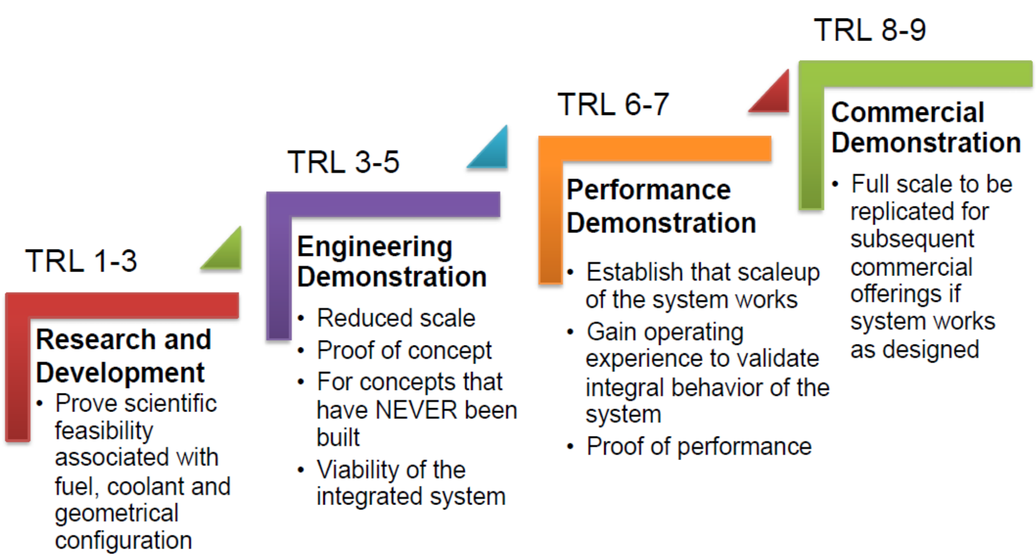

Figure 3-1
Representation of Reactor Developments Steps. [Image courtesy of US Department of Energy, 2016].

3.1 Background on the Government-Private Sector Initiatives in the Commercialization of Nuclear Power

For the purpose of this study, the following definitions apply:

Test (or Experimental) Reactors – with little or no electric power generation, used for reactor physics investigations, materials testing, safety research, feasibility studies, and often training and process development. With some exceptions, these were predominantly built by the involved government, often with private sector involvement in design, engineering, and construction. Those that preceded the first demonstration reactors (definition follows) can be thought of as precursor test reactors, as they laid the initial foundation for a decision to invest in scaled-up projects. Follow-on testing exists to continuously increase technical understanding and explore ways to improve commercial and safety performance.

Small Demonstration Reactors – built to support the goal of commercializing a technology, but not necessarily to be cost-competitive themselves. These reactors all had greater private sector involvement than the Test Reactors that preceded them.

Large Demonstration Reactors – designed and built to fully demonstrate long-term, significant power generation at powers that were about 40-50% of final commercial reactor sizing. Substantial private sector involvement and/or leadership with reduced government funding and involvement was the norm.

Commercial Reactors – reactors that were built by the private sector, in accordance with the country's final commercialization strategy.

3.1.1 Policy to Commercialize Nuclear Energy

As documented by the DOE Office of History and Heritage Resources [DOE 1983]: "Almost a year after World War II ended, Congress established the United States Atomic Energy Commission (AEC) to foster and control the peacetime development of atomic science and technology. Reflecting America's postwar optimism, Congress declared that atomic energy should be employed not only in the Nation's defense, but also to promote world peace, improve public welfare, and strengthen free competition in private enterprise." This mission was enabled by the signing of the Atomic Energy Act in 1946, and the AEC presided over this mission as well as the ongoing nuclear defense mission. In the service of these missions, the AEC organized the nation's centers of atomic research and development into the National Laboratory network that provided support to the government's production facilities as well as R&D relevant to future nuclear power applications and other peace-time applications.

In nuclear military and national security missions, the U.S. government had long established the practice of employing domestic private industry for a full range of services such as: parts fabrication, architect-engineering, construction, installation, and facility operations. Therefore, the practice of government and industry interfacing in the service of cutting edge technological endeavors was not new in the sphere of atomic technology. Under the AEC, as early as 1947, familiar corporate names were involved with the National Laboratories in construction, fabrication, and installation of nuclear test reactors – the nascent practical demonstrations that the heat produced by a nuclear reactor could be used to generate electricity. Similarly, the government had demonstrated the viability of nuclear submarine propulsion by 1953. Therefore, by this time, a limited slice of domestic private industry had been exposed to the depth of government-owned atomic technology and R&D capabilities. It was the idea of private investment and private ownership, however that was new.

3.1.2 Industrial Participation Program

By 1950, domestic U.S. industrial leaders were expressing interest in understanding nuclear technology. At this time, the AEC considered the possibility of developing a dual-purpose reactor that could be employed in plutonium production as well as in the production of electrical power. In 1951, consistent with both its national security and its domestic commercialization missions (a concept that would be more fully developed by the United Kingdom's nuclear program), the AEC responded to industry interest by launching the Industrial Participation Program. This program granted four industry teams access to information necessary to perform scoping studies on R&D, design, and development of such a dual-purpose reactor. Although the project was not pursued when it was realized that such additional plutonium production capability was no longer needed, the effort proved valuable to the AEC's commercialization mission. Importantly, the AEC learned that the high costs of applied R&D, as well as capital equipment and structures, were a barrier to private industry which would require cooperative government-industry arrangements to address. Additionally, the government realized that the traditional practice of remotely locating atomic installations was not practical for delivery of electricity to the public. Therefore, government needed to address new issues for nuclear power plant siting. The AEC also recognized a need to appropriately ease certain security restrictions on industry access to relevant nuclear technology. More support in policy and practices were going to be needed to launch the AEC's commercialization mission.

3.1.3 A Five Year Plan

In the next development, as owner of the technical bases for nuclear technology development, government was well-positioned to identify potential concepts for power reactor R&D, and did so in 1954 with a plan to build 5 experimental reactor concepts in 5 years. The government established the concepts of interest as follows and initiated government-owned test reactor projects [NSF 1977a, AEC 1972].

- Pressurized water reactor,
- Sodium graphite reactor,
- Boiling water reactor,
- Fast breeder reactor, and
- Homogeneous reactor.

Notably, these projects included the prototype PWR at Shippingport, PA and the Argonne Experimental Boiling Water Reactor, foreshadowing the dominance of these technologies in the U.S. commercial nuclear power market.

3.1.4 Improvements in Policy Implementation

Further commercialization support, however, came with the Eisenhower administration when the President's well-known 1953 Atoms for Peace proposal before the United Nations became U.S. national policy. Necessary details of policy administration came in 1954 with modifications to the Atomic Energy Act (AEA) which were designed to stimulate and incentivize industrial commercialization of nuclear power.   The AEA modifications enabled the AEC to launch the

Power Demonstration Reactor Program (PDRP) in January of 1955. The Program was launched in 4 phases known as: the first round, the second round, the third round, and the modified third round. Each round solicited proposals from private industry for the development and implementation of nuclear power reactor concepts.

3.1.5 PDRP – First Round

Under the first round of PDRP solicitations, the AEC indicated that the following types of government assistance to proposers would be considered [JCAE 1964]:

- Waiver of established AEC charges for loan of source and special nuclear materials, up to an agreed ceiling for a period of 7 years from July 1, 1955. Participants would be required to pay for any consumption of materials and for services performed by the AEC such as recovery of materials from spent fuel elements.
- Performance of agreed upon research and development work in AEC laboratories, without charge to participants.

The criteria used in AEC evaluation of proposals included:

- Estimated probability of proposed concept to achieve economically competitive power production,
- Cost to the AEC in funds and materials,
- Risk assumed by the proposer,
- Competence and responsibility (financial strength) of the proposer, and
- Assurances by the proposer against abandonment of the project.

The AEC received four proposals in the first round of the PDRP: the Yankee Atomic Electric Company, which proposed a PWR design (Yankee Rowe); the Power Reactor Development Company, which proposed a sodium-cooled fast reactor (Enrico Fermi); and the Consumers Public Power District, which proposed the sodium-cooled graphite-moderated thermal reactor (Hallam Nuclear Power Facility); and the Nuclear Power Group (NPG) headed by Commonwealth Edison Company, which proposed a BWR design (Dresden Unit 1). The Commonwealth Edison proposal converted to an independent project, while the remaining three projects proceeded as First Round PDRP projects. All four reactor project were carried out to completion [AEC 1972].

3.1.6 PDRP – Second Round

The invitation to the second round of the PDRP was announced on September 21, 1955 seeking the development of small, economically competitive plants to be built by small public utilities with an upper bound of 40 MWe capacity. A number of proposals were submitted and, four reactor facilities were constructed and operated as the result of the Second Round – the Elk River BWR in Elk River Minnesota, the Piqua Nuclear Power Facility (organic cooled and moderated) in Piqua, Ohio, the Boiling Reactor Nuclear Superheat Project (BONUS) in Punta Higuera, Puerto Rico, and the La Crosse BWR in Genoa, Wisconsin [NSF 1977a, AEC 1972, AEC 1964]. Notable factors for the second round include [JCAE 1964]:

- Continuation of the assistance available of the First Round, but expansion of assistance to include AEC financing of all or part of the construction of power reactors and the retention of title to the portion of the plant paid for by the AEC.
- Similar proposal evaluation criteria, however, the AEC made explicit that the technological advancement represented by the proposed design was to be considered.

3.1.7 PDRP – Third Round

The third round of the PDRP was announced in January 1957 and required proposals by December 1958 (later extended to June 1959). No limitations were placed on the types or sizes of plants which could be proposed, except that "they should make significant contributions toward the achievement of commercial utilization of nuclear power and that construction will be completed by June 30, 1962" [JCAE 1963]. Consistent with the earlier Five Year Plan, a goal of the Third Round was design diversity. The AEC was particularly interested in developing a heavy water reactor (HWR) that would utilize natural uranium fuel, and also a homogeneous reactor [JCAE 1954]. The AEC received a design proposal for the HWR from the Carolinas-Virginia Nuclear Power Associates [IAEA 2002, Morris 1992]. Other proposals accepted as part of the Third Round were: Northern States Power proposed the Pathfinder BWR in South Dakota; Consumers Power Company proposed the Big Rock BWR in Big Rock Point, Michigan; Philadelphia Electric proposed the Peach Bottom high-temperature gas reactor, and California Edison and Westinghouse proposed the San Onofre PWR [NSF 1977b]. All of these projects proceeded to construction and operation. Consistent with AEC's desire to see industry initiative in homogeneous reactor concepts, an independent project was considered by Pennsylvania Power & Light and Westinghouse, but the project did not advance to construction [AEC 1972].

The types of assistance available in the third invitation were characterized as having "essentially duplicated the arrangement first used at Shippingport…the AEC owns the reactor, bears related research and development costs, and bears the cost of fabricating the first core. The utility owns the electrical portion of the plant and operates the complete facility" [Davis 2014a]. The three main types of government assistance offered were the following [Oulahan 1959, Stason 1959]:

- Waiver of established AEC charges for use of source and special nuclear materials over a specified period of time (the free use of heavy water was offered for the first five years of plant operation for plants designed to be fueled with natural uranium)
- Performance in AEC laboratories of agreed-upon R&D which could not be reasonably performed elsewhere. Done at no cost or at less than full cost.
- Provision of R&D support that is necessary to advance the technology of projects which promise to make a significant contribution toward achieving inexpensive, abundant, and safe nuclear power.

The criteria for acceptance of proposals were:

- Cost to the AEC in funds, materials, and services
- Probable contribution of the proposed project towards achieving economically competitive power in a plant of the size proposed
- Technical and financial competence of the applicants

3.1.8 The Modified Third Round

In August of 1962, the AEC announced the continuation of the third round of the PRDP, which came to be known as the Modified Third Round. Proposals were solicited for the design, construction, and operation of nuclear plants with at least a 400 MWe capacity [AEC 1989]. The major objective of this round was the demonstration of design, construction, and operation of large nuclear power plants which were reliable sources of electric power under baseload conditions. Third Round types of assistance (discussed above) were offered. Additionally, the AEC offered design assistance for the complete plant.

Two proposals were received and contract negotiations were completed in 1963. One proposal was from Connecticut Yankee Atomic Power Company, a corporation sponsored by a group of New England utilities, and the other was from the Department of Water and Power of the City of Los Angeles. The latter proposal did not advance past the design phase. The Connecticut Yankee project became the Haddam Neck PWR plant [AEC 1964].

3.1.9 Establishing a National Policy on Utility Liability

In the midst of the implementation of the PDRP, the federal government enacted the Price-Anderson Act of 1957. As detailed by NRC historians Mazuzan and Walker [Mazuzan 1984], this piece of legislation was two years in the making beginning with congressional concern that the nascent U.S. commercial atomic power business was not growing fast enough to meet the national agenda or to keep pace with other countries that were embracing the peaceful application of atomic technology. Hearings with U.S. utility executives surfaced industry's discomfort with shouldering the full burden of nuclear accident liability. Leaders in the private insurance industry who were called by Congress to study nuclear power plant liability concluded that the nuclear power industry represented an unprecedented circumstance in at least two respects. First, there was insufficient actuarial data for large nuclear power facilities on which to establish a basis for risk and necessary rates. Second, the implications of an extreme accident scenario implied potential hazard and damage beyond the boundaries of the nuclear facility and, therefore, potential for extensive additional liabilities. Notably, commercial facilities would be located closer to population centers than those nuclear facilities utilized for national defense missions. The insurance industry did, however, make significant efforts to form industry syndicates to spread the risk and to define a peak value that their industry was willing and able to insure. The insurance industry predicated its underwriting of the commercial nuclear industry, however, on the formation of inspection programs and the existence of government safety oversight [Mazuzan 1984].

Nonetheless, the level of underwriting offered by private insurance was deemed insufficient by U.S. utilities. The utilities concluded that federal assistance was needed to fill the gap, and the concept of accident liability indemnification materialized. In order to validate the need for such a measure, Congress directed Brookhaven Laboratory to answer such fundamental questions as: What is the probability of a fission product release? What sort of situation would result in distribution and exposure of fission products to members of the public? What sort of exposure or contamination would cause harm to the public and damage to property? And, finally, what would be a worst case consequence in terms of human injuries, fatalities, and property damage? The results were documented in WASH-740, "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants," which would be the first of many studies sponsored by the federal government to understand and strengthen nuclear power plant safety

[AEC 1957]. Brookhaven's results, although couched in terms that emphasized significant technological uncertainty in the science of accident characterization, amply convinced Congress that private insurance coverage had to be augmented in some way by federal legislation. In fact, General Electric told Congress that if indemnification legislation did not pass, work on Commonwealth Edison's Dresden Station would stop [Mazuzan 1984]. President Eisenhower signed the bill into law on September 2, 1957.

What began as an inquiry into the basis for utility liability insurance evolved into the beginnings of structured safety analysis for the benefit of public health and safety. The question of utility liability forced a rendering of what was known and what was not known about the potential for and consequences of commercial nuclear power accidents. This influenced the course of subsequent federal R&D, establishing the government's technical role not only in the development of technologies, but also the technical bases for the safety of those technologies – through the continuation of reactor test programs and technical safety studies to this day. Additionally, specific terms added by Congress to the Price Anderson Act set a new tone for a regulatory approach that centered around public safety and stakeholder involvement. Specifically, Congress included a provision to introduce a degree of separation between that part of the government that was promoting the nuclear industry and that part that was overseeing its safety. Two further provisions were made - one regarding public access to safety-related reports and one regarding public hearings on reactor projects [H.R. 1414 2016]. Mazuzan and Walker document GE's response to the passage of Price Anderson as: "…truly significant milestone and sets the stage for continued, rapid industrial progress" [Mazuzan 1984].

3.1.10 Summary of a 3-Pronged Initiative to Support Commercialization in U.S.

The initiative to launch a thriving commercial nuclear power industry in the U.S. can be viewed in three parts. One involves federal policy. Another involves technological development and collaboration between government and industry. And, a third involves management of financial burden and risk. The history of U.S. commercial nuclear power ascendency is rooted in federal policy. The Executive and Legislative branches of government strongly supported the objective of the peaceful extension of atomic science. The policy manifested itself in the construction of test facilities and the conduct of seminal R&D, with the contracted help of private industry. A concerted transition from high-security technology development to commercial technology development and deployment needed additional policy in the form of legislation, however – the 1954 modifications to the Atomic Energy Act. The legislation allowed the federal government to engage directly with private industry to incentivize private enterprise in the design, construction, and operation of nuclear power facilities.

Some utilities, such as the smaller utilities in more rural areas which responded to PDRP Round 2, needed more financial assistance than others. Other larger utilities were eager to take full financial control of early reactor projects. Yet, the utilities at both end of the spectrum relied upon the foundations of atomic science laid by the federal government, both as part of defense initiatives and as part of R&D specific to nuclear power. Industry projects also benefited from federal investment in fuel enrichment processes – something that could not have been accomplished by any one utility on its own. Utilities that participated in the PDRP benefited from a range of terms that lessened financial burdens such as those associated with acquisition of nuclear materials, conduct of technology-relevant R&D, and project construction. Whether part of the PDRP or not, all utilities, after 1957, benefited from the Price Anderson Act's indemnification from nuclear accident damages – a provision that exists today in a modified form.

The last PDRP reactor, San Onofre, came on-line in 1968. By that point in time, there were completed projects that were wholly funded by private industry (Dresden 1, Indian Point 1), and also a number of projects well into construction that were wholly funded by private industry (Oyster Creek and Quad Cities BWRS, and RE Ginna and HB Robinson PWRs). Although PWR and BWR technology has matured and dominated the U.S. commercial power market and nuclear utilities have managed themselves independently for some time, government policy, technology development, and management of financial risk continue to influence the current commercial nuclear power industry and its possible paths forward. Government sponsorships and incentives for improvements in existing technologies and government involvement in the development of new technologies are occurring today. Further, the Price Anderson Act continues to exist today, in amended form. Therefore, what was accomplished in the era of the PDRP had established early principles for how government and private enterprise interact in the commercial nuclear industry. Lasting principles about policy, technology development, and risk management may inform future paths for the next advancements in nuclear power delivery

3.1.11 Canada and UK Technology Selections

As with the U.S., the Canadian and UK governments amassed understanding and experience with atomic science as the result of their WWII endeavors and alliances. Canada and the UK were wartime members of the Manhattan Project, and the UK had its own nuclear research body [AEC 1989, Whitlock 2016b, DOE 1962]. Thus, the UK and Canadian governments were well-positioned to explore the technical viability of peaceful applications in atomic science, and both governments made it policy to do so. In parallel with LWR development in the U.S., the UK developed a gas-cooled power reactor (GCR) technology, and Canada developed a heavy water power reactor (HWR) technology. As in the U.S., these technology development paths began with R&D activities initiated and directed by government, with limited contracted private sector assistance, followed by small and large demonstration reactors with industry involvement, and ultimately culminated in the launching of full technology commercialization embraced and implemented by industry.

After WWII, Canada developed a path to peaceful nuclear applications tuned to its domestic resources – namely, its vast amount of natural uranium and the supply of heavy water inventory from Eurpose that was in Canada's possession. As early as 1950, a reactor based on natural uranium and heavy water was advocated by the government's Chalk River Laboratories. In 1953, Chalk River hosted the first Atomic Energy Canadian Laboratory (AECL) power symposium, which was attended by industry and utility representatives. The symposium reflected Canadian policy to focus the government (AECL) role on R&D support and to require early industry and utility involvement. In November 1953, the power program began to move forward, and by 1954, utilities, engineering companies, and manufacturers were invited to participate in a feasibility study of nuclear power. Representatives from nine industry partners convened at Chalk River to form the Nuclear Power Group (NPG). The NPG functioned as a unit of Chalk River's engineering design division and was led by Ontario Hydro – a private utility. The aim of the group was to develop a nuclear power reactor that would be economically competitive in the electrical power market. Ontario served as the locus of engineering and scientific expertise in nuclear energy, and served as the industrial base for production of nuclear fuel [Whitlock 2016b].

The UK government had considerable depth in nuclear technology due to the nation's war-time involvement in the Manhattan Project as well as the formation of its own committee of expert university researchers [Whitlock 2016b, DOE 1962]. With the passage of the Atomic Energy Act in the U.S. in 1946, collaboration in nuclear technology with the UK was reduced, leading to a decision by the UK government to pursue an independent UK nuclear weapons program [AEC 1972, Sneddon 2010]. Starting in 1946, nuclear research in the UK was focused on the production of plutonium for nuclear weapons applications. By the mid-1950s, government policy expanded the application of UK atomic science to also pursue the production of civilian electrical power. This shaped the choice of the technological approach to power reactor design, leading to the gas-cooled power reactor concept and the fleet of MAGNOX civilian power reactors.

3.2 US Pressurized Water Reactors (PWR) Development

The early success of the PWR development is often attributed to several defense-related projects and test reactor facilities; two of the most pertinent are described in the following sections. Next, the first PWR demonstration project, Shippingport Atomic Power Station, is discussed. It marks the beginning of the 15 year lead time to the start-up of the first commercial-scale PWR that was fully funded by industry. Construction and operation of test reactors did not end at this point, however. Government and private industry continued to investigate reactor phenomena with ongoing testing in parallel with construction of the first power reactors. Sequential PWR demonstrations, increasing in size and capacity, are discussed culminating in the first commercial-scale PWR constructed without government assistance, RE Ginna in New York [Davis 2011a].

Precursor Test Reactors:

- Materials Testing Reactor (MTR) in Arco, ID
- Submarine Thermal Reactor (STR) Mark I in Arco, ID

First PWR Demonstration Project:

- Shippingport Atomic Power Station (SAPS) in Shippingport, PA near Pittsburg

Subsequent Test Reactors:

- Special Power Excursion Reactor (SPERT-3) in Arco, ID
- Westinghouse Test Reactor (WTR) in Waltz Mill, PA
- Transient Reactor Test Facility (TREAT) in Arco, ID
- Saxton Nuclear Experimental Reactor Project in Saxton, PA
- Loss of Fluid Test Facility (LOFT) in Arco, ID

Successive PWR Demonstration Projects and Full-Scale Power Plant:

- Yankee Rowe in Rowe, MA
- Indian Point in Buchanan, NY
- Haddam Neck near Hartford, CT
- San Onofre in Pendleton, CA
- RE Ginna in Ontario, NY

3.2.1 Material Testing Reactor (MTR) – Test Reactor

The need for a multi-purpose test reactor available for testing and materials development was identified as early as 1944 [DOE 1962, ANS 2004]. What became the Materials Test Reactor (MTR) was designed jointly by Oak Ridge National Laboratory (ORNL) and Argonne National Laboratory (ANL). The MTR was a 30 MWth water-cooled and water-moderated reactor using enriched uranium-235 fuel and beryllium as a reflector located at the National Reactor Testing Station (NRTS, now the Idaho National Laboratory, INL) in Arco, Idaho [AEC 1972, Davis 2012]. The Blaw-Knox Corporation acted as architect-engineer, and the plant was built by the Fluor Corporation. Personnel from Phillips Petroleum Company were trained at ORNL to operate the MTR and the reactor became operational in 1952 [Davis 2012, ORNL 1952]. Because of its high specific power and high neutron flux, the MTR could be used to test samples of structural material candidates that needed to withstand the high flux environments that were planned for commercial scale reactors. The MTR operated successfully as one of the most highly demanded test reactors, performing over 15,000 irradiation experiments during its 18-year operational lifetime [ANS 2004]. The MTR has a place in nuclear history in the United States as the first general purpose test reactor.

Not only did the MTR provide irradiation testing services, but the plant itself was a precursor demonstration of the light water reactor technology, and the design of the MTR contributed directly to the naval nuclear propulsion project [AEC 1989]. Other contributions by the MTR were that fuel for the first BWR Experiment (BORAX) was constructed from slightly modified MTR fuel subassemblies, and the MTR support systems also supplied the de-mineralized water to fill the BORAX reactor tank [Haroldsen 2008]. Therefore, the MTR highly benefited the development of both the PWR and BWR reactor technologies. The MTR was defueled in August 1970, and parts of the facility were used for other purposes and the DOE made the decision in 2005 to dispose of the facility [Davis 2012].

3.2.2 Submarine Thermal Reactor (STR) Mark I and Mark II – Test Reactor

In December 1948, the AEC contracted with Westinghouse to operate the Bettis Atomic Power Laboratory (Bettis), where the submarine propulsion plant was designed and developed in coordination with reactor design work performed at ANL--thus establishing the general relationship under which ANL and Westinghouse would follow in designing and developing components for the Mark I and NAUTILUS. Extensive exposure of material samples in Hanford and ORNL reactors, along with the NRX test reactor in Canada, confirmed the selection of zirconium as a cladding material to be used for Mark I—a change from aluminum cladding used in the Hanford production reactors and the MTR. ORNL had been successful in devising a process for separating zirconium from hafnium, but the production of large quantities of acceptably pure zirconium was still uncertain; major efforts by the Foote Mineral Company and

Bettis were required to perfect the process and produce the zirconium for the Mark I. Continued research on improved processes, along with the declassification of the technology after passage of the 1954 amendments to the AEA, assisted commercialization of the production of zirconium by 1955 [Hewlett 1974, AEC 1972].

A timetable of the STR Mark I project is found below [DOE 1983, Davis 2014b, DOE 2011]:

- April 1948 – Formal project established at ANL
- June 1948 – First Navy-Westinghouse contract
- December 1948 – Original AEC–Westinghouse contract
- March 1949 – AEC announces election of a site in Idaho for the NRTS
- March 1950 – Westinghouse occupancy of new facilities at the Bettis Site (outside Pittsburgh, PA)
- August 1950 – Commencement of Mark I construction at NRTS
- August 1951 – Award of Mark II construction contract to Electric Boat Division, General Dynamics Corporation
- June 1952 – Keel plate laying of USS NAUTILUS (SSN 571)
- March 1953 – First critical operation of Mark I prototype plant
- June 1953 – Mark I full design power reached
- January 1954 – Launching of USS NAUTILUS
- September 1954 – Commissioning of USS NAUTILUS
- March 1980 – Decommissioning of USS NAUTILUS
- October 1989 – Permanent Shutdown of Mark I

3.2.3 Shippingport Atomic Power Station (SAPS) – Small Demonstration Reactor

In a December 7, 1953 press release, the AEC invited industry to submit proposals to partner to build a nuclear reactor for generating electric power. This was an opportunity for industry to obtain firsthand experience with the new technology involved in building and operating a large-scale reactor designed specifically for power-producing purposes [AEC 1989]. As a result of this solicitation, nine proposals were received [JCAE 1963]. Subsequently, the AEC entered into a contract with Duquesne Power & Light (DP&L), effective March 18, 1954, for the accomplishment of the project known as the Shippingport Atomic Power Station (SAPS). DP&L provided the site for the nuclear power plant at Shippingport, PA. DP&L also constructed and operated a new turbogenerator plant and ancillary facilities at the new site, and operated the entire plant after training provided by Bettis [Hewlett 1974]. DP&L assumed up to $5 million of the costs connected with R&D of the reactor portion of the plant and purchased steam generated by the reactor at specified rates (rates that represented a 30-40 percent premium over the price paid from coal-fired plants) for the first five years of operation [AEC 1989]. Westinghouse Electric Corporation was selected as the AEC's principal contractor for the design, development, and construction of the reactor portion of the plant. The Stone & Webster Engineering Corporation was awarded a subcontract by Westinghouse to perform architect-engineering services for the reactor part of the plant. The Dravo Corporation was selected as the subcontractor to install the nuclear portion of the SAPS plant, to perform shop fabrication work, and to construct certain associated facilities, while DP&L selected Burns & Roe to perform architect-engineering services for the non-nuclear portion of the plant [Hewlett 1974, JCAE 1955].

SAPS was built in thirty-two months at a reported total cost of approximately $73 million in 1957 dollars [JCAE 1974, NRC 2014a]. This was below projected costs from the Joint Committee on Atomic Energy (JCAE), which had estimated in 1954 that the construction cost of the plant, exclusive of the turbine generator portion, would be approximately $38 million and estimated R&D costs of $44 million, totaling nearly $82 million in 1957 dollars [JCAE 1963]. Most importantly, the AEC estimated that three to four years would be needed to bring SAPS online, and their schedule was on target. From the initial invitation announcement on December 7, 1953, barely four years had passed when SAPS started generating electricity for commercial use on December 18, 1957 [NRC 2014a].

Shippingport Atomic Power Station (SAPS) was the first large-scale, civilian nuclear power plant. It was designed to be of moderate power (60 MWe) and was a conversion from a defense project for a naval aircraft carrier reactor. SAPS marked the first nuclear reactor facility designed specifically to support peaceful uses of nuclear energy and was a centerpiece of the Eisenhower Administration's "Atoms for Peace" program. Because of its importance to U.S. national policy at the time, it helped set the course for light water reactors to dominate the U.S. market. It is difficult to overstate the importance of SAPS in the context of commercializing the LWR, and particularly the PWR. The AEC understood that this initial large-scale demonstration would not necessarily be cost-effective; however, they believed that the PWR was a proven, available design that could be used both as an important part of "Atoms for Peace" and as a tool for the AEC and industry to gain construction and operating experience with a near full-scale plant. It was an opportunity for the U.S. to amass data and experience that could only be obtained from an operating facility – data such as plant availability, equipment reliability, period of amortization, and operating and maintenance costs [JCAE 1954]. In summary, the government built and owned the reactor, incurred the majority of the nuclear R&D costs, as well as the cost of fabricating the first core of fuel. DP&L owned the electrical portion of the plant and operated the entire facility under AEC supervision [Morris 1992, AEC 1957].

3.2.4 Yankee Rowe – Large Demonstration Reactor [PDRP]

During the first round of the PDRP, the AEC received four proposals. One was from the Yankee Atomic Electric Company (YAEC) of Boston, Massachusetts; their proposal was accepted by the AEC and a contract was awarded on June 4, 1956. YEAC was a corporation created and owned by a large consortium of New England utility companies [JCAE 1955]. In addition to the incentives available through the PDRP, such an industry partnership was a way to spread risk and move forward with innovative projects.

The contract provided that YAEC would build and operate a PWR design of at least 134 MWe (392 MWth) in generating capacity. Yankee Rowe was constructed with a capacity of 175 MWe and the YAEC consortium and its subcontractors – comprised of developers, suppliers, and users – were to bear all the financial risks associated with building and operating the reactor and power generating facility. YAEC contracted with Westinghouse Electric Corporation for the development and design of the nuclear plant, and with Stone & Webster for architect-engineering and construction services. It was estimated that construction costs would total $34.5million, of which $18.5 million (1955 dollars) would be for the reactor and for equipment associated with the reactor portion of the plant. All design and construction costs would be borne by YAEC [JCAE 1963].

Three kinds of government assistance were to be provided by the AEC [NSF 1977b]:

- Fuel use charges would be waived for initial years of operation,
- An agreed share of preconstruction R&D work would be performed in national laboratories without charge, and
- R&D aspects of post-construction operations would be subsidized (by law the AEC could not fund the actual construction of reactors to be used commercially).

It was estimated that research and development costs would total $5 million. Under the contract, the AEC agreed to perform up to $1 million of R&D in its own facilities and to reimburse, up to $4 million, the cost of R&D work performed in private facilities. All R&D costs in excess of $5 million were to be borne by YAEC. In addition, the AEC agreed to waive its normal charge for the use of special nuclear material to fuel the reactor, amounting to approximately $3.3 million for the first 5 years of operation of the plant. The upfront costs included in the initial contract summed to approximately $44 million in 1956 dollars [JCAE 1963].

The YAEC PWR project was successful, possibly the most successful of the PDRP reactors, at least from a cost and schedule standpoint. It was completed on time and under budget by $5 million, at an estimated total cost of $39 million, and began operating in 1960 [JCAE 1974]. Yankee Rowe was permanently shut down in 1991 [NRC 2014b]. A key feature of this project was that the financial risks were borne by a consortium of a dozen utilities plus the additional manufacturing companies who joined forces. Yankee Rowe's success was also subsequently attributed to the fortunate circumstance that PWR technology was advanced sufficiently for demonstration due to previous government-led RD&D [NSF 1977b].

3.2.5 Indian Point – Large Demonstration Reactor

The Indian Point nuclear plant of Consolidated Edison was considered significant in the development and commercialization of the PWR because government assistance was not used to finance the facility construction due to expectations of economic competitiveness with coal in the northeast at that time [JCAE 1956]. Cost projections by the AEC totaled to approximately $128 million in 1973 dollars for Indian Point [JCAE 1974]. Located in Buchanan, New York, Indian Point was a 265 MWe (615 MWth) reactor that began commercial operations in August 1962 [ORNL 1973, Lovering 2016]. First core operations used thorium and highly enriched uranium provided by the AEC, but then a switch was made to lightly enriched uranium for the remainder of the reactor's operational lifetime [JCAE 1965]. The plant was shut down in October 1974 because the emergency core cooling system did not meet evolving regulatory requirements [NRC 2015a].

The following Industry Partners were involved with the Indian Point project [ORNL 1973]:

- Owner Utility: Consolidated Edison Company of New York
- Architect/Engineering Firm: Consolidated Edison Company
- Reactor Pressure Vessel Vendor: Babcock & Wilcox
- Nuclear steam supply Vendor: Babcock & Wilcox

3.2.6 Haddam Beck – Large Demonstration Reactor [PDRP]

In response to the Modified Third Round of the PDRP, the Connecticut Yankee Atomic Power Company proposed the Haddam Neck nuclear power plant [JCAE 1963]. Connecticut Yankee Atomic Power Company represented a consortium of New England utilities, which requested AEC assistance in connection with the design, construction, and operation of a larger PWR design. The AEC and Connecticut Yankee Atomic Power Company began negotiations considering a 490 MWe capacity plant, but soon thereafter, many components for the Connecticut Yankee plant were sized to 590 MWe providing the potential to increase the generating capacity. The final design capacity was around 580 MWe [JCAE 1963, Lovering 2016]. The plant location was about 20 miles southeast of Hartford, Connecticut. Industry involvement included Stone & Webster as responsible for the overall design, scheduling, procurement, and construction of the plant, and Westinghouse Electric Corporation was the reactor vendor [JCAE 1963]. Combustion Engineering was the reactor designer [ORNL 1973]. The estimated capital cost at the beginning of the project was around $85 million in 1964 dollars for the Connecticut Yankee project, but increased to $99 million dollars in 1974 [JCAE 1963, JCAE 1974]. AEC assistance would be limited to reimbursement for early reactor plant design, up to a ceiling of $6 million, and, in addition, the AEC would waive nuclear material use charges up to $7.1 million [JCAE 1963]. Haddam Neck began commercial operation in January 1968 and was closed in 1996 [Gammell 2016].

3.2.7 San Onofre – Large Demonstration Reactor [PDRP]

In November 1960, between PDRP Round 3 and the Modified 3rd Round, the AEC received an unsolicited proposal from the Southern California Edison Company and the Westinghouse Electric Corporation for the design, development, construction, and operation of a PWR design with an initial design capacity of 395 MWe [JCAE 1963] (which was constructed to 430 MWe [JCAE 1974]). The San Diego Gas & Electric Company later joined the proposed arrangement. The AEC reached agreement with the companies on the principal features of this proposal, but the project was still in need of a site. Agreement was reached that the most feasible site for the plant, to be known as the San Onofre Nuclear Generating Station (SONGS), would be on the Marine Corps reservation at Camp Pendleton, California. However, legislation was necessary in order to authorize the Secretary of the Navy to grant long-term easements to the utilities for the use of this site. Conditioned upon the enactment of such legislation and the consequent grant of the required easements, the AEC moved forward.

Although San Onofre was financed with government assistance, it can be viewed as the beginning of the “Turnkey” era in nuclear power when reactor vendors offered fixed-cost contracts for designing, constructing, testing, and meeting regulatory requirements [Burness 1980]. Southern California Edison contracted with Westinghouse and the Bechtel Corporation to procure the nuclear plant which Westinghouse-Bechtel jointly designed and constructed. The initial estimated capital cost of the plant in the early 1960s was approximately $84 million. Southern California and San Diego were responsible to the AEC for the construction and operation of the plant for 5 years. The AEC waived fuel use charges up to a ceiling of $6.5 million. The AEC also contracted with Westinghouse for an R&D program for which the AEC would reimburse Westinghouse for costs up to $6.4 million. In July 1963, the Navy was authorized by Congress to grant the necessary easements [JCAE 1963]. The total cost of the project was later estimated to be $97 million in 1968 dollars [JCAE 1974], which equated to over

85% of the financing being provided by private industry [NSF 1977b]. San Onofre started commercial operations 1968 and was shut down in 1992 [NRC 2016a].

3.2.8 RE Ginna – Initial Comercial Reactor [PDRP]

The RE Ginna nuclear power plant (Ginna) has received less attention in the literature of the history of LWRs; however it is significant as being the first large-scale commercial plant fully built without government support. It has been described elsewhere that San Onofre was the milestone that achieved the AEC's original goals outlined to Congress [NSF 1977a, NSF 1977b]. However, because government assistance was provided at San Onofre, Ginna is probably more appropriately recognized as the first commercial PWR plant. Ginna was also part of the "Turnkey era" in which reactor vendors would design, construct, test, apply for licensure, and hand over the "key" to the owner [Burness 1980]. Located in Ontario, New York (about 20 miles northeast of Rochester, New York) [NRC 2016b], Ginna's construction began April 1966 and commercial operations started in December 1969 [EIA 2011]. The plant was designed and operated at a capacity of 496 MWe and was uprated in 2006 to allow for a 580 MWe capacity [ORNL 1973, NEI 2016, Exelon 2015]. The operational license runs through September 2029 [Exelon 2015].

The following industry partners were involved with the RE Ginna PWR project [ORNL 1973]:

- Owner Utility: Rochester Gas & Electric Company
- Operator: R.E. Ginna Nuclear Power Plant, LLC
- Architect/Engineering Firm: Gilbert Associates
- Reactor Pressure Vessel Vendor: Babcock & Wilcox
- Nuclear Steam Supply System (NSSS) Vendor: Westinghouse
- Containment Constructor: Bechtel Corporation

3.2.9 Subsequent Government and Private Testing Facilities

The RE Ginna nuclear power plant (Ginna) has received less attention in the literature of the history of LWRs; however it is significant as being the first large-scale commercial plant fully built without government support. It has been described elsewhere that San Onofre was the milestone that achieved.

SPERT-III, LOFT, and TREAT

In addition to the precursor test facilities built and operated by the U.S. government, additional government-sponsored test reactors continued to be planned, designed, constructed, and operated in the interests of furthering technological development and understanding of transient phenomena. Three are mentioned here. The conceptual design for the AEC's Special Power and Excursion Reactor (SPERT-III), located at the NRTS, began in 1955, and the completed facility operated from 1958 through 1968. SPERT-III was designed to incorporate nuclear and hydraulic features relevant to PWR and BWR technology and was used to investigate reactor response to challenging variations in operational parameters such as temperature, pressure, and flow [AEC 1969]. The AEC's Transient Reactor Test Facility (TREAT), also located at NRTS, operated between 1959 and 1994. TREAT was air-cooled, graphite-moderated, thermal spectrum reactor designed to simulate a broad range of accident conditions in order to test prototypic fuel pins and fuel bundles. This capability is so essential to the understanding of fuel behavior that the DOE is considering the use of TREAT for current research into concepts for accident tolerant fuels [INL 2013]. Similarly, the AEC's Loss of Fluid Test facility (LOFT), which was in the planning stages in the mid-1960s and operated between 1973 and 1985 at NRTS, was used to simulate PWR response to postulated loss-of-coolant-accidents (LOCAs) [INL 1979]. The many SPERT and LOFT experiments were important to the early and ongoing development and validation of computational modeling used in commercial nuclear power reactor design and licensing. SPERT, LOFT, and TREAT exemplify the early and continuous role of the U.S. government in nuclear power reactor technology development.

WTR and Saxton Nuclear Experimental Reactor

Private industry was also motivated to invest in testing facilities during the initial and ongoing growth of U.S. PWR commercialization. Westinghouse designed, built, and operated its own materials testing facility, fashioned after the AEC's MTR but designed to 60MWt versus 40MWt. As with the MTR, its purpose was to study the effects of radiation on fuels and equipment. The Westinghouse Test Reactor (WTR) operated from 1959-1962 at the Westinghouse Waltz Mill, PA research site [ORNL 1963]. Westinghouse was also the designer for a 20 MWth single-loop PWR test reactor connected to the power generating side of the coal-fired Saxton Generating Station near Altoona, PA. This facility was sponsored by the Saxton Nuclear Experiment Corporation, which was formed by a consortium of four northeastern utility companies. The facility operated from 1962 through 1972 and performed experiments on the use of boron for reactivity control; it also experimented with mixed-oxide (MOX) fuel [Davis 2013].

3.3 US Boiling Water Reactors (BWR) Development

Government-initiated projects and experiments of the BWR concept are presented in the following sections. As mentioned above, it should be noted that BWR work benefitted from much of basic and applied research done previously to support PWRs [NSF 1977a]. Two major test reactor programs were run for BWR technology in national laboratories, and later private industry also built test facilities a test reactor. Next, three separate major large-scale BWR demonstrations are discussed. These efforts culminated in the first fully commercial PWR project, Oyster Creek [NSF 1977b].

Government-funded test reactors:

- Boiling water reactor experiments (BORAX I-V) at Arco, ID
- Experimental Boiling Water Reactor (EBWR) at ANL, near Chicago, IL

Privately-funded test reactors:

- General Electric Testing Reactor (GETR), Pleasanton, CA
- Vallecitos Boiling Water Reactor (VBWR), Pleasanton, CA

Demonstration projects and first commercial plant:

- Dresden in Morris, IL
- Elk River in Elk River, MN

- Big Rock Point in Big Rock Point, MI
- Humboldt Bay Unit 3 in Humboldt Bay, CA
- Boiling Reactor Nuclear Super Heat Project, Punta Higuere, Puerto Rico
- Pathfinder Atomic Power Plant, Sioux Falls, SD
- La Crosse Boiling Water Reactor (LACBWR) in Genoa, W
- Oyster Creek near Toms River in NJ.

3.3.1 Boiling Water Reactor Experiment (BORAX) – Test Reactors

The BORAX projects were all under the direction of the ANL. Five experimental reactors were involved in the overall project. The first, a boiling reactor experiment known as BORAX-I, was assembled and operated at the NRTS in 1952-1953; it provided a basis for confidence in proceeding with development of boiling water reactors. Fuel for the first BORAX reactor made use of slightly modified fuel subassemblies from the Material Testing Reactor (MTR). The MTR was already in operation nearby at NRTS. It was mid-summer of 1953 before BORAX-I was ready to start operation. De-mineralized water was transported from the MTR facility to fill the reactor tank [Haroldsen 2008].

BORAX-I was intended to determine the feasibility and stability of the reactor if boiling occurred within the core. It was purposely subjected to severe transient testing and finally tested to total destruction on July 22, 1954 [Haroldsen 2008]. Further experiments were conducted due to the promise that BORAX-I had shown [Haroldsen 2008]. BORAX-II was designed, built, and operated by ANL to replace BORAX-I and to investigate a new reactor configuration that would more closely approximate the characteristics of a practical power reactor. BORAX-II was constructed during the summer of 1954 and went critical October 1954. Tests of new core combinations were tried using varying enrichments of uranium-235 in the metal fuel plate at power levels around 6 MWth [ANL 2016a].

Plans were made to build BORAX-III even before the final tests of BORAX-II were completed. While earlier versions of the BORAX experiments were conducted outdoors in the open air, BORAX-III was contained within the shelter of a sheet metal building. On July 17, 1955, for two hours, the 6 MWth BORAX-III reactor generated approximately 2000 kW of electricity, with 500 kW used to power the BORAX facility, 1000 kW used to power the Central Facilities Area at NRTS, and 500 kW to light the city of Arco, Idaho [ANL 2016a]. BORAX-III is remembered as the nuclear power plant that made history on when it provided the first electrical power to an American town (Arco, Idaho, west of Idaho Falls) [DOE 2011].

Work and testing of several variations of BORAX continued after BORAX-III with BORAX-IV and BORAX-V. BORAX-IV was a 20 MWth (2.5 MWe) reactor used to test thorium-uranium ceramic fuel and later was purposefully refueled with defective fuel elements to develop techniques for locating failed fuel elements within the core during operations. BORAX-IV first went critical in December 1956 and was shutdown June of 1958 [Haroldsen 2008]. The final experiment in the BORAX series, BORAX-V, was operated from 1962 to August 1964. The main objective of the BORAX-V experiments was to test nuclear superheating concepts and the stability of the BWR design at high power densities. A superheater was added in two separate configurations: a centrally located superheater within the core and a superheater located on the peripheral of the reactor. The latter configuration was successful and proved that higher cycle efficiency of heat transfer to produce steam could be achieved. Similar to the BORAX-IV experiment, BORAX-V was operated then intentionally with defective fuel elements, but this time to test if significant contamination would occur in the power conversion equipment with the use of a superheater [Haroldsen 2008].

Information derived from BORAX-I through BORAX-III were of great value in the design and operation of the experimental boiling water reactor (EBWR), as described in the next section. Subsequent development of the commercial-scaled BWR design also benefitted from information from the BORAX experiments [JCAE 1955].

3.3.2 Experimental Boiling Water Reactor (EBWR) – Small Demonstration Reactor

The Experimental Boiling Water Reactor (EBWR) was designed, constructed, and operated at ANL. It was intended to serve two missions. First, it was intended to provided data related to the design and operation of a potential commercial BWR reactor design. Second, it was also intended to provide valuable data to the military for evaluating a BWR as a possible design for small scale power units (the Army SL-1 design) [AEC 1989]. The EBWR was designed to produce 20 MWth (5 MWe) and operated with an enriched uranium core [Boing 1991]. Later in its life, operations were converted to plutonium to support the AEC's Plutonium Recycle Program [JCAE 1954, ANL 2016a, Boing 1991].

The EBWR plant was designed at a power level that was believed to be the minimum capacity which enabled sound extrapolation to large size central station power plants (i.e., engineering scale) [JCAE 1955]. Both the EBWR and its precursor, BORAX-III, demonstrated that a BWR design was feasible without serious radioactive contamination of the steam turbine [ANL 2006]. The architect-engineer for the EBWR building was selected and a fixed-price contract awarded to Sargent & Lundy of Chicago, Illinois. Allis-Chalmers Manufacturing Company was awarded a fixed-price contract for the design, construction, and installation of the power equipment [JCAE 1955]. Graver Tank & Manufacturing fabricated the reactor containment structure; while Babcock & Wilcox Company designed and fabricated the reactor pressure vessel [ANL 1957]. EBWR program was closed in 1962. The facility was decommissioned in 1996 [Boing 1991, ANL 2006].

AEC's experimental BWR programs came to an end in the mid-1960s. The General Electric Company had taken the initiative, was performing its own technical development at its test reactor at Vallecitos facility in California (discussed below), and was actively marketing its BWR designs (Dresden Nuclear Power Station). They developed and marketed full-sized commercial boiling water reactor power plants based, in part, on the technical information developed through the BORAX experiments and the EBWR [Haroldsen 2008].

3.3.3 Vallecitos Boiling Water Reactor (VBWR) & General Electric Testing Reactor (GETR) – Test Reactor

The Vallecitos Boiling Water Reactor (VBWR), located near Pleasanton, California, was the first privately funded and constructed nuclear power plant to supply power in megawatt amounts to an electric utility grid. Because the VBWR served as a test reactor for the Dresden BWR project, it was designed with a high degree of operating flexibility for testing of various aspects of boiling water reactor operation, nuclear stability, alternative control systems, instrumentation, and heat transfer. The VBWR plan was approved by General Electric's management in late 1955, and was issued the first Power Reactor License by the AEC at a capacity of 5 MWe (40 MWth) [ASME 1987]. In addition, VBWR was a training facility for engineers, physicists, supervisors and operating personnel on the new BWR technological system. Construction began in June 1956 and was completed in 1957. Pacific Gas and Electric Company (PG&E) installed and operated the turbine generator. The turbine was a standard General Electric marine unit that was modified to accept saturated steam [ASME 1987]. The reactor fuel was composed of highly enriched uranium-235 metal sandwiched fuel between stainless steel cladding [ASME 1987, JCAE 1957a]. The VBWR was estimated at $4 million at the beginning of the project, but was completed under budget and on schedule [ASME 1987, JCAE 1957a].

The VBWR first went critical on August 3, 1957. It was connected with the utility grid on October 19, 1957 and was shut down on December 9, 1963 [ASME 1987]. During VBWR's lifetime, many changes were made to its structure and operating procedures. License amendments increased the power levels at which it was permitted to operate. Experiments that were carried out included the following [ASME 1987]:

- Dresden prototype element irradiations;
- Fuel element irradiations for post-irradiation heat transfer analyses based on microscopic examination;
- Metal specimen irradiations to determine the effects of fast neutron radiation on fracture and tensile properties;
- Fuel specimen irradiations to study the fission gas release phenomenon
- First known nuclear production of superheated steam in the Superheat Advanced Demonstration Experiment (SADE); and
- Irradiations to study neutron reactions (transmutations).

VBWR was on the campus of the Vallecitos Nuclear Center (VNC) that also operated the General Electric Test Reactor along with hot cells, fuel, chemistry and metallographic laboratories [Morrissey 1979a]. The GETR was a materials test reactor that operated from December 1958 through October 1977. Its capabilities and missions were similar to those of the government's MTR and TREAT facilities. GE's commercial fuel testing program conducted fuel rod testing over a wide range of power conditions [Morrissey 1979b]. A research reactor onsite and hot cells to perform post-irradiation examination are still in active use today [NRC 2016c].

3.3.4 Dresden – Large Demonstration Reactor

Dresden was a 200 MWe (630 MWth) BWR in Morris, Illinois designed by General Electric; Bechtel Corporation was contracted as the architect/engineering firm on the project [NSF 1977a, JCAE 1963, Davis 2011b]. The Nuclear Power Group, which was a joint venture headed by Commonwealth Edison Company with six other utilities, made history when Dresden Unit 1 was the first nuclear power plant to be financed entirely with private funding [NSF 1977a, JCAE 1965] at an estimated cost of $117 million in 1955 dollars [JCAE 1974]. Dresden Unit 1 was also significant because of the technological extrapolation that was required by the General Electric Company to utilize the knowledge gained through the BORAX experiments, the EBWR, and the VBWR to design a large-demonstration BWR [Haroldsen 2008]. It supported plans by

General Electrical to ". . . progress from experimental through demonstrations to "target' and eventually "large' BWR plants." With limited operating experience available, GE had designed a significant capacity increase in building the first multi-hundred MWe installation [NSF 1977b].

The construction of Dresden Unit 1 began March 1, 1957 and the plant reached initial criticality in October 1959. Commercial power production commenced in June 1960 [Davis 2011b]. The unit experienced minor steam leaks and erosion in steam piping in the early and mid-1960s. Dresden Unit 1 experienced fuel failures in the late months of 1964 that led to redistribution of radionuclides from the fuel to other parts of the primary system. None of the fuel failures caused releases above the regulatory limits, and Dresden Unit 1 produced power on the grid until October 31, 1978 [NRC 2016d]. This early shutdown date occurred because Dresden was temporarily shut down for maintenance when the Three-Mile Island incident occurred in March 1979. Problems with Dresden arose when commercial operations began because this was the initial integration of larger-scaled systems required to produce large amounts of electric power. The1979 shutdown became permanent due to the challenges posed in meeting new regulatory expectations [NRC 2016d].

3.3.5 Elk River – Small Demonstration Reactor (PDRP)

Elk River was acclaimed in the early 1960's as "rural America's first atomic power plant" by the Rural Cooperative Power Association (RCPA), and the reactor was intended to supply power to RCPA customers in and around Elk River, Minnesota [Davis 2011a]. Elk River was selected during the second round of the PRDP in February of 1956 as a basis for negotiation to construct a 22 MWe closed-cycle boiling water power reactor (CCBR). The second round PRDP stipulations were developed after the AEC concluded (after the first round of the PRDP negotiations) that there was sufficient need and potential application of nuclear power, up to a 40 MWe range, to call for cooperative efforts in the development of small economically competitive nuclear power plants [JCAE 1963, AEC 1989]. Elk River was also to be the first thorium-fueled boiling water power reactor and would use an oil-fired boiler to produce super-heated steam. Further, the project involved the AEC teaming with the RCPA; it was, therefore, of unique technological and programmatic interest to the AEC and Congress [JCAE 1963, AEC 1989, JCAE 1957b].

The initial negotiations between the AEC and RCPA agreed that RCPA would provide the site and turbogenerating facilities and would operate the entire plant for five years as part of its system. Under a separate agreement with the AEC, the plant was to be constructed by American Machine & Foundry (AMF) Company. Both direct contracts with the AEC requested that the bids indicate a maximum ceiling cost to the AEC [JCAE 1963, JCAE 1957b]. Additionally, RCPA would purchase steam from the AEC, as well as provide the operating staff of the entire plant once turned over from the reactor vendor, AMF. The original concept outlined by RCPA stated that the plant would be completed in late 1959 for a total cost of $6.2 million, but during the final bidding, cost estimations grew and were revised to $8 million, with no fixed ceiling, and a completion scheduled for 1961 [JCAE 1963, NSF 1977b].

Re-negotiations between the AEC and RCPA ensued; in March 1958, the AEC selected the proposal of the Nuclear Products-Erco Division of ACF Industries, Washington, D.C., as a basis for contract negotiation to design, develop, construct, and test-operate the Elk River reactor for the RCPA and AEC [JCAE 1963]. On May 2, 1958, agreement was reached with ACF under which ACF would design, fabricate, and construct the nuclear plants (reactor and superheater), train RCPA personnel, and test-operate the nuclear plant. This work was to be performed under a cost-type contract with a maximum cost ceiling to the AEC of $9.3 million including a fixed fee. Under the agreed arrangements, RCPA was to provide the site and the turbogenerator facilities, estimated to cost $1.75 million. RCPA would also operate the entire plant for five years as part of its system and would buy all the steam produced by the reactor plant [JCAE 1963]. ACF engaged Sargent & Lundy (Chicago) as the architect-engineer firm for the project, and the Maxon Construction Company of Dayton Ohio as its general contractor. ACF projected completion in 1960 at a cost of $11.4 million [Davis 2011a].

Construction of the Elk River plant began January 1959 and reached initial criticality in November 1962 [IAEA 2016]. Elk River provided electricity first in August 1963 while full commercial operations started July 1964. However, Elk River experienced technical and economic challenges and operated only for the five year contracted period. It was shut down in February 1968 [NSF 1977a, NSF 1977ab].

3.3.6 Big Rock Point – Small Demonstration Reactor (PDRP)

An unsolicited proposal from the Consumers Power Company (CPC) of Jackson, Michigan, and the General Electric Company was submitted to the AEC on December 18, 1959 to build a 75 MWe BWR. The proposal was styled after the third round of the PRDP. The proposal requested AEC assistance under third round terms in connection with the design, development, construction, and operation of a high power density boiling water reactor to be located at Big Rock Point, Michigan [JCAE 1974].

Under the contract signed on January 5, 1961, CPC, at its own expense, was to design, construct, and operate the nuclear plant for five years. The AEC would reimburse CPC for post-construction research and development costs up to $500,000 and would waive fuel use charges up to a maximum of $1.65 million for the first five years of operation. Under a separate contract with General Electric, the AEC would support a program of pre- and post-construction R&D up to a maximum cost of $3.7 million (subject to 10 % escalation). CPC estimated that its capital costs for the plant would be $27.8 million. CPC's principal contractors were Bechtel, for the design and construction of the plant, and General Electric Company for the supply of reactor fuel [JCAE 1963]. Overall, the AEC provided about 20% of the funds for Big Rock Point which was typical of the support provided for the third round projects [NSF 1977b]. Big Rock Point was constructed in 29 months as a 70 MWe BWR (scaled down slightly smaller than the original bid at 75 MWe) and on budget [JCAE 1974, Tompkins 2006].

Criticality was reached on September 27, 1962. Commercial operations began November 1, 1965, and continued for 32 years until shutdown in 1997. From 1969 to 1977, Big Rock Point was also licensed to use mixed-oxide fuel through a cooperative R&D program that included GE, Exxon, and CPC and was sponsored by the Edison Electric Institute (now the Electric Power Research Institute, EPRI) [Tompkins 2006]. Site remediation was completed on August 29, 2006 [NRC 2016e].

3.3.7 Humboldt Bat – Small Demonstration Reactor

The Pacific Gas & Electric Company (PG&E) had participated with General Electric and Bechtel to privately fund and build the VBWR test reactor (discussed above) and PG&E had provided the turbine and distributed the small amount of generated power. The experience with the VBWR boosted the utility's interest in privately funding and operating its own nuclear power generating unit. On the basis of an economic analysis performed for a new power source at PG&E's Humboldt Bay site near Eureka, CA, PG&E announced its intent to build a 60 MWe BWR power plant to augment the two existing fossil fuel units in 1958 [Herbert 2012]. Two favorable economic factors cited at the time were the use of the ocean as a heat sink versus building cooling towers, and the comparatively modest fuel transportation charges due to the once-a-year fuel reloading schedule.

Humboldt Bay Unit 3 was designed by PG&E and General Electric. The design incorporated a pressure suppression system for accident mitigation. The design of the pressure suppression system would subsequently be adapted to other BWR designs. Construction began in 1960. Combustion Engineering furnished the reactor vessel, while GE provided the turbine and generator. Bechtel was the construction contractor. Construction was completed and operations began in 1963. The government provided enriched nuclear material that was manufactured into BWR fuel by GE. In 1976, after 13 years of operations, the facility shut down for a planned refueling and a year-long $30M seismic upgrade. However, geological surveys discovered a nearby active fault, indicating the need for more seismic studies. With the passage of time, PG&E realized that an additional investment of at least $300M would be necessary to implement post-TMI safety improvements [Herbert 2012]. Under these circumstances, PG&E the decision was made to shutdown Unit 3 permanently.

3.3.8 Integral Nuclear Superheat BWRs – Small Demonstration Reactors

As discussed above, the AEC sponsored the BORAX series of test reactor experiments under the direction of ANL to investigate and support the development of BWR technology. The fifth and last experiment in the series, BORAX-V, took place between 1962 and 1964 and concerned the concept of designing a portion of the reactor core to impart additional heat to steam originally developed by a different part of the core. The objective is to transform saturated steam into superheated to increase power generation efficiency. Two configurations were tested. Positive results are reported for a configuration where the superheating would occur on the periphery of the core; however, some planned tests regarding safety considerations were not performed [ANL 2016a]. Interestingly, prior to the start of the BORAX-V experiment, the AEC accepted two PDRP proposals for Integral Nuclear Superheat BWR demonstration reactors.

As part of Round 2 of the PDRP, the AEC entered into agreement with the Puerto Rico Water Resources Authority (PRWRA) to design, build, and operate the BONUS reactor (Boiling Nuclear Superheat reactor). The AEC agreed to fund the nuclear portion of the plant and the PRWRA agreed to fund the balance of plant and to provide the operating staff. The facility was designed by General Nuclear Engineering Corporation. The design incorporated four peripheral nuclear superheaters and was intended to be capable of producing steam at 900^{o}F and produce 16.3 MWe [AEC 1964, Davis 2016]. Construction was begun in 1960 (2 years before BORAX-V testing) and first criticality was achieved in 1964. Initial operations were conducted in boiling water mode, slowly incorporating the superheating and reaching full power in 1965. However, the project encountered problems due to: a complex design, missteps in procurement, and shortfalls in design implementation. Due to the high expected costs to resolve these problems, a decision was made to terminate the program in 1968 [Davis 2016].

Nearly in parallel with BONUS, the AEC accepted a Northern States Power (NSP) PDRP Round 3 proposals for a second Integral Nuclear Superheat BWR – the Pathfinder Atomic Power Plant located in Sioux Falls, South Dakota. In this arrangement, NSP committed more than $30M to construct the facility. A utility consortium contributed $3.65M towards associated R&D, while the AEC contributed $8M. Further, the AEC agreed to waive fuel charges for the first five years of operation, which equated to $1.8M [Davis 2014c]. In this design by the Allis-Chalmers Manufacturing Company, the nuclear superheater was located in the center of the core (which had been evaluated in the BORAX-V tests, but had not operated as favorably as the peripheral design incorporated into BONUS). The planned power level of Pathfinder was 58.5 MWe. Construction commenced in 1959 (several years before BORAX-V testing) and criticality was achieved in 1964. Two years were spent in low-power testing and adjustments before reaching 90% power for a short time in 1967. Inspections performed after this point revealed internal component failures, fuel cladding erosion, and leakage resulting in secondary side contamination. The problems were deemed to be too severe and uneconomical to solve, so NSP make the decision to terminate the project in 1968. The nuclear steam supply system was permanently isolated and the balance of plant was converted to a fossil fuel facility [Davis 2016].

It is interesting to observe that the timeline of events that comprises the history of development of a superheated system design (BORAX-V, BONUS, and Pathfinder) deviated, to a degree, from the stepped progression of technology development evidenced by the overall history of BWR and PWR commercial nuclear power. First, the small demonstration reactor designs and start of construction for BONUS and Pathfinder preceded relevant BORAX testing (although significant BWR-related testing had been completed at BORAX previously and was on-going at EBWR). Second, significant investment was made in two parallel prototype projects at practically the same time, removing the possibility of lessons learned to be passed from one project to the other. Third, some planned BORAX testing was never completed.

3.3.9 La Crosse – Small Demonstration Reactors (PDRP)

The AEC received an unsolicited proposal in April 1961 from the Dairyland Power Cooperative (DPC) and the Allis-Chalmers Manufacturing Company to build and operate a 50 MWe BWR in Genoa, Wisconsin, called the La Crosse BWR (LACBWR). The AEC, DPC, and Allis-Chalmers reached an initial agreement quickly after, but had to rework the conditions of indemnification and warranties of performance according concerns from the Rural Electrification Administration (REA, a division of the Department of Agriculture [NARA 2016]) for providing funding for the LACBWR. Outlines of the REA conditions were such that the AEC was to reimburse DPC for damages and special systems costs incurred by the utility in the event of delays in reactor completion, or in the event of a failure of the reactor to perform in accordance with the design specifications; DPC would pay liquidated damages to the AEC for delays in completion of the turbo-generator. REA then agreed to provision of $6.7 million to DPC for its costs associated with the project. Under the 1962 contract with the AEC, and based on second round PDRP conditions, DPC would assume the cost of the conventional facilities, estimated at around $7.9 million, and operate the plant for 10 years. The AEC would finance the design and construction of the reactor, which it would own, reimburse DPC for operating costs, and sell steam generated by the reactor plant to DPC for 10 years. DPC, in addition, would bear the cost of the turbogenerator and balance of plant equipment. Under a separate contract with the AEC, Allis-Chalmers agreed to design and construct the reactor, train operating personnel, and perform preoperational testing for the fixed price of $11 million [NRC 2015b]. The Allis-Chalmers

Company was the original licensee; the AEC later sold the plant to DPC and provided them with an operating license. Total cost to the AEC was estimated at $16.3 million [JCAE 1963]. The plant was completed in 1967 and operated until April 1987 [Lipa 2012].

3.3.10 Oyster Creek – Initial Commercial Reactor

One of the more significant milestones in civilian power development occurred on December 12, 1963, when the Jersey Central Power and Light (JCPL) Company announced that it had contracted for a large nuclear power reactor to be built at Oyster Creek near Toms River, New Jersey. According to the company's evaluation, the plant would be competitive with a fossil fuel plant. For the first time, an American utility company had selected a nuclear power plant on purely economic grounds without government assistance and in direct competition with a fossil-fuel plant [DOE 1983]. From 1963 to 1969, design and planning were conducted for Oyster Creek by JCPL and industry partners [NSF 1977b]. Oyster Creek went online December 1969 at a capacity of 637 MWe and at a capital cost ranging from $91-96 million in 1969 dollars for construction [JCAE 1974, NJ Department of Planning 2014]. Oyster Creek is licensed to operate until 2029, but is presently scheduled to shut down in 2019 [Wald 2010].

The following Industry Partners were involved with the Oyster Creek project [ORNL 1973]:

- Owner Utility: Jersey Central Power & Light Company
- Architect/Engineering Firm: Burns & Roe
- Reactor Pressure Vessel Vendor: Combustion Engineering
- Nuclear steam supply: General Electric
- Containment Constructor: Chicago Bridge & Iron

Initial acceptance in the utility marketplace, starting with the Oyster Creek plant, was also stimulated by vendor offers to construct and guarantee the operability of the plants for firm fixed prices – the “turnkey plant contracts” [NSF 1977b, Burness 1980]. Thus, the Oyster Creek operation start date of December 1969 marks the date that full commercialization had begun under open market terms similar to those that utilities had long favored in purchasing fossil fuel plants in earnest [NSF 1977b, SF 1977a].

3.4 Summary of Government and Industry Roles in Early LWR Demonstrations

A summary of the types of assistance from the government for the reactors described in this paper is documented in Table 3-1 and Table 3-2.

Table 3-1
Government and Industry Roles in PWR RD&D.

Reactor, [Dates of Operation], Location, Electricity Generation Capacity	**Government Role**	**Industry Role**
LWR Materials Testing Reactor (MTR) [1952-1970] National Reactor Testing Station (NRTS) in Arco, ID 30 MWth	• R&D: Pre-construction • Designer (ANL, ORNL) • Owner • R&D: Post-construction • Fuel Fabricator & Supplier	• Blaw-Knox*: Architect/Engineer (A/E) firm • Fluor Corporation*: Constructor • Phillips Petroleum Company*: Operator *Under contract with AEC, no ownership
PWR Submarine Thermal Reactor (STR) Mark I [1953-1989] NRTS in Arco, ID 10 MWe	• R&D: Pre-construction • Conceptual Designer (ANL) • Owner • Operator (Bettis Atomic Power Laboratory) • R&D: Post-construction • Fuel Testing (ORNL, Hanford), Fabricator, and Supplier	• Westinghouse*: Contractor Operator of the Bettis Atomic Power Laboratory, Reactor Designer, Limited Fuel Testing *Under contract with AEC, no ownership

PWR
Shippingport Atomic Power Station (SAPS) 1st & 2nd Cores
[1957-1974]
Shippingport, PA
60 MWe
	• R&D: Pre-construction
	• Reactor Owner
	• R&D and Construction of Reactor past $5M
	• Supplier of the 1st Core fuel loading
	• Waived 1st Core use charges,
		• Duquesne Light Co.:
		  ○ Site owner
		  ○ Reactor Operator
		  ○ ≤$5M R&D and Construction of Reactor
		  ○ Turbogenerator Constructor and Operator
		  ○ Steam Customer at above-market rates
		• Westinghouse: Principal Reactor Designer, Developer & Constructor
		• Stone & Webster: Reactor A/E firm
		• Dravo Corp.: Reactor Installation

PWR
Yankee Rowe
[1960-1991]
Rowe, MA
175 MWe
	• R&D: ≤$5M Pre- and Post-construction R&D
	• Reactor Owner
	• R&D: Post-Construction
	• Waived 1st Core use charges for first 7 years
		• Yankee Atomic Electric Co.:
		  ○ Any R&D required excess of $5M
		  ○ Reactor Operator
		  ○ Turbogenerator Owner and Operator
		• Westinghouse: Principal Reactor Designer, Developer & Constructor
		• Stone & Webster: Reactor A/E firm

PWR
Indian Point
[1962-1974]
Buchanan, NY
265 MWe
	• No Government Financial Assistance
	• Participated in Thorium fuel cycle R&D with the AEC
		• Consolidated Edison Company of New York
		  ○ Site Acquisition and Owner, Operator
		  ○ A-E firm
		• Babcock & Wilcox:
		  ○ Reactor designer

Table 3-1 (continued)
Government and Industry Roles in PWR RD&D.

Reactor, [Dates of Operation], Location, Electricity Generation Capacity	Government Role	Industry Role
PWR Haddam Neck [1968-1996] Near Hartford, CT 580 MWe	• Reimbursement for titles I and II design, (up to $6M) • Waive fuel use charges up to $7.1M	• Connecticut Yankee Atomic Power Company: ○ Site Acquisition and Owner, Operator • Stone & Webster: ○ Overall plant designer, responsible for scheduling, procurement and construction • Westinghouse Electric: Reactor Vendor • Combustion Engineering: Reactor Designer
PWR San Onofre Unit 1 [1968-1992] Pendleton, CA 430 MWe	• R&D: ≤$6.4M reimbursed to Westinghouse • Reactor Owner • Waived fuel use up to $6.5M	• Southern California Edison Co.: ○ Site Acquisition and Owner ○ Reactor Operator for 5 years ○ Turbogenerator Owner and Operator • Westinghouse-Bechtel-San Diego Gas & Electric Partnership: Reactor Designer, Developer & Constructor • Westinghouse: $6.4M of R&D
PWR RE Ginna [1969-Present] Ontario, NY 580 MWe	• No Government Assistance	• Rochester Gas & Electric Company ○ Site Acquisition and Owner • RE Ginna Nuclear Power Plant LLC ○ Operator • Gilbert Associates: A-E firm • Westinghouse Electric: Reactor Vendor • Babcock & Wilcox: Reactor Designer • Bechtel Corporation: Containment constructor

Table 3-2
Government and Industry Roles in BWR RD&D.

Reactor, **[Dates of Operation], Location, Electricity Generation Capacity**	**Government Role (AEC, unless noted)**	**Industry Role**
BWR BORAX I-V National Reactor Testing Station [NRTS] in Arco ID [1953-1964] BORAX III: 2 MWe	• R&D: Pre-construction (ANL) • Designer and Constructor (ANL) • Owner • R&D: Post-construction (NRTS/INL) • Fuel Fabricator & Supplier	• Phillips Petroleum Company*: NRTS operating site contractor *Under contract with AEC, no ownership
BWR Experimental Boiling Water Reactor (EBWR) ANL near Chicago, IL [1956-1967] 5 MWe, later increased to 100 MWe	• R&D: Pre- and post-construction • Reactor Designer (ANL), Operator (ANL), Owner	• Allis-Chalmers Manufacturing Co.: ○ Designer and Constructor/Installer, of the power equipment • Sargent & Lundy: Reactor building A/E firm • Babcock & Wilcox: Designed and fabricated reactor pressure vessel • Graver Tank & Manufacturing: fabricated reactor containment
BWR Vallecitos BWR (VBWR) near Pleasonton, CA [1957-1963] 40 MWth; 5 MWe	• No Government Assistance (**)	• General Electric: ○ Designer, Owner, Operator • Pacific Gas and Electric Company (PG&E): ○ Installed and operated the turbine generator
BWR Dresden Morris, IL [1960-1978] 200 MWe	• No Government Assistance (**)	• Nuclear Power Group: ○ Owner, Operator • General Electric: Reactor Designer • Bechtel Corp.: A/E firm
BWR Elk River Elk River, MN [1963-1968] 22 MWe	• Reactor Owner • Waived initial fuel use charges	• Rural Cooperative Power Associates: ○ Site owner ○ Reactor Operator ○ Turbogenerator Owner and Operator ○ Steam Customer • ACF Industries: ○ Design, fabricate, and construct the nuclear plants (reactor and superheater), ○ Train RCPA personnel, ○ Initial testing of initial operations of the reactor plant • Sargent & Lundy: Facility A/E firm • Maxon Construction Co.: General Contractor

Table 3-2 (continued)
Government and Industry Roles in BWR RD&D.

Reactor, [Dates of Operation], Location, Electricity Generation Capacity	Government Role (AEC, unless noted)	Industry Role

BWR
Big Rock Point
Big Rock Point, MI
[1965-1997]
70 MWe
	• R&D:
	  o ≤$3.7M pre- and post-construction support to General Electric subject to 10% escalation
	  o ≤$500k post-construction R&D reimbursed to Consumers
	• Reactor Owner
	• Waived fuel use up to $1.65M for the first 5 years of operation
		• Consumers Power Company:
		  o Site owner
		  o Design, construct, and operate the nuclear plant for five years
		  o Reactor Operator
		  o Turbogenerator Owner and Operator
		  o Steam Customer
		• General Electric:
		  o Reactor designer and R&D
		  o Supplier of reactor fuel
		• Bechtel: design and construction of the plant subcontractor

BWR
La Crossee Boiling Water Reactor (LACBWR)
Genoa, WI
[1967-1987]
50 MWe
	• AEC:
	  o Original Owner*
	  o Financed design and construction
	  o Reimburse DPC operating costs
	  o Agreement to DPC and REA conditions: Damages, delays to reactor and turbogenerator costs reimbursed back to REA
	• Rural Electrification Administration (REA [a division of the Department of Agriculture]: provided up to $6.7M to support the overall project
		• Dairyland Power Cooperative (DPC):
		  o Site acquisition, Operator and eventual Owner*
		  o Purchase steam from AEC for 10 years
		  o Purchase turbogenerator and balance of plant equipment
		• Allis-Chalmers:
		  o Original plant licensee
		  o Reactor designer and constructor
		  o Trained operating personnel
		  o Perform pre-operational testing

BWR Oyster Creek near Toms River, NJ [1969-Present] 637 MWe	• No Government Assistance	• Jersey Central Power and Light Company: Owner, Operator • Burns & Roe: A-E firm • Combustion Engineering: Reactor designer • General Electric: Reactor vendor • Chicago Bridge & Iron: Containment Constructor

3.5 Canada's RD&D to Commercialize CANDU

3.5.1 HWR Research and Test Reactors – Test Reactors

Three research reactors were the basis of proving the basic technical features of the HWR: Zero Energy Experimental Pile (ZEEP), National Research Experimental (NRX) reactor, and National Research Universal (NRU) reactor. A detailed proposal for ZEEP was developed in 1944 as a part of the Manhattan Project. ZEEP was designed to operate at 1 watt (thermal); it was fueled with natural uranium and used heavy water as a moderator. The first criticality was reached on September 5, 1945 [AECL 1997].

The NRX project was a 10 MWth heavy-water-moderated, natural-uranium fueled reactor that was used for research and produced plutonium-239 and irradiated thorium samples to produce uranium-233. Irradiations of NRX fuel were successful at demonstrating that the total heat output could compete economically with fossil-fuel with prospects for longer irradiations. AECL performed the necessary R&D and carried most of the cost of the nuclear steam generating system. The Canadian General Electric Company (CGE) created their Civilian Atomic Power Department (CGE-CAPD) and designed the nuclear steam generating system, oversaw the plant construction, and undertook manufacturing of certain components. Ontario Hydro provided the site, designed the balance of plant, paid for the balance of plant, and was to commission and operate the station. Ontario Hydro was willing to reimburse AECL for all energy produced at a competitive rate based on current market prices from coal generation. Contracts were solicited from private industries for the design and supply of most of the plant components [McConnell 2002]. In 1951, NRX began producing molybedenum-99 [Brown 2009]. Based on NRX's successful operation at 10MWth and improved understanding, NRX was operating at 40 MWth by 1954 [AECL 1997]. On December 12, 1952, a partial meltdown of the aluminum calandria core occurred; however, NRX was placed back in service within two years in 1954 and permanent shutdown of the NRX reactor occurred in 1992 [Brown 2009].

The NRU design was developed by the National Research Council of Canada at Chalk River; its operation was later taken over by the AECL. NRU and was based on the NRX design but now incorporated means for continuous operation with on-line refueling while also confirming that high-power output was attainable from a relatively small reactor. However, NRU was designed initially to 200 MWe to support production of electricity, but later in the design stage was reconfigured to 135 MWth to produce plutonium for sale, carry out R&D, and supply medical radioisotopes [CNL 2016]. It is also noted that the US Navy testing at the NRX provided verification that a high-temperature, high-pressure water system with zirconium cladded nuclear fuel could operate practically [Hewlett 1974, AECL 1997]. NRU went critical for the first time on November 3, 1957 and presently produces 40% of the world's supply of Molydenum-99 and a significant amount of cobalt-60 for medical uses. NRU is scheduled to shut down in 2018 [MacLeod 2015, CNL 2015].

3.5.2 Nuclear Power Demonstration (NPD) – Small Demonstration Reactor

The Nuclear Power Demonstration (NPD) served as the proof-of-concept for the HWR technology and was a small demonstration that built on the success of the three predecessor test reactors (ZEEP, NRX, and NRU). By the end of 1954, it was concluded by NPG that a power rating of 20 MWe for NPD-1 was technically feasible. Once the design parameters for the prototype were agreed upon, proposals were invited for detailed design and construction. Of the seven bids received, AECL selected Canadian General Electric (CGE) in March 1955 as the prime contractor for NPD-1. CGE would contribute $2 million in engineering development in support of the design within their Civilian Atomic Power Division (CAPD). The initial total cost of NPD-1 was expected to range between $8-20 million [Whitlock 2016, McConnell 2002]. Ontario Hydro was confirmed as the utility to be involved in the project and would supply the site and the conventional power equipment, buy steam from the nuclear system, and operate the plant. AECL would provide funding and own the nuclear system of NPD-1 [AECL 1997].

In 1958, the AECL and Ontario Hydro had determined that the original NPD-1 design would not be constructed, and the NPD-1 project was canceled in the design phase. Focus would turn to the NPD-2 design and would target 1961 as the NPD-2 commission date. The NPD-1 conceptual design was an earlier iteration of the reactor design with a pressure-vessel design, similar to LWRs, while the NPD-2 used a pressure-tube design, along with a horizontal calandria and the online refueling capability that would become distinctive features of CANDU reactors [Cantello 2003, Bereznai 2011]. NPD-2 was also a testbed for new fuel-bundle designs, alternative pressure-tube materials, new instrumentation, and other components.

During the 20 MWe NPD-2 design study, AECL established the Nuclear Power Plant Division (NPPD) in Toronto (early in 1958) in space provided by Ontario Hydro. Ontario Hydro would be the ultimate customer and agreed to participate in the design study and to supply supporting engineering personnel. The construction was performed by Canadian Bechtel Ltd. under the direction of CGE. During the NPD-2 project, Ontario Hydro recruited key operating staff, formed a rigorous training program, and developed the operational procedures that became the pattern for all Ontario Hydro nuclear stations and other CANDU operators [McConnell 2002]. Cost reviews of NPD-2 estimated that the total price was approximately $34 million in 1955 Canadian dollars [McConnell 2002].

Initial operation of NPD-2 occurred in April 1962, and full power was reached on June 28, 1962. NPD-2 was located near Rolphton, Ontario and successfully demonstrated the feasibility of the HWR concept through reliable operation, safe refueling at full power, and acceptably low heavy water losses. The 20 MWe NPD-2 operated from 1962 to 1987 and provided training opportunities for designers, researchers and operators that would then manage the growth of the HWR program over the next two decades [AECL 1997]. In 1985, it became evident that its pressure tubes should be replaced. After a prolonged shutdown, it was decided that the cost of re-tubing NPD-2 would exceed the value of continuing to operate, and it was taken out of service in 1987 [AECL 1997].

3.5.3 Douglas Point – Large Demonstration Reactor

Douglas Point signified the first large-scale demonstration of the HWR technology, and it was during its development that the term "CANDU" was coined (CANada Deuterium-Uranium reactor). During this design effort, Ontario Hydro and the AECL shared design and development tasks. By the spring of 1959, the government, AECL, and Ontario Hydro were ready to start work on the 200 MWe Douglas Point facility because of the rapid pace of development of nuclear power observed internationally. Canada's Federal Cabinet approved the Douglas Point project in June 1959, and land near Kincardine on Lake Huron's shoreline was acquired. Construction began in February 1960. The agreement between AECL and Ontario Hydro outlined that AECL would design and own the station, and Ontario Hydro would construct and operate the facility and pay for the electricity. Ontario Hydro would also own the site and have an option to buy the station once it had demonstrated successful operation. The Douglas Point project established the organizational pattern for Ontario Hydro's nuclear power program. It was a learning experience for NPPD, Ontario Hydro's construction division, and a large number of Canadian manufacturers [AECL 1997]. Douglas Point did not produce electricity as cheaply as expected, and Ontario Hydro never exercised its option to buy the station from the AECL. However, revenue earned by Ontario Hydro paid for most direct operating costs [AECL 1997].

The estimated project cost was $81.5 million, with a target date of 1965 for full operation. A number of first-of-a-kind issues delayed the construction of Douglas Point. The Douglas Point project was stretched out to seven years by delays in completing the design and by late deliveries of equipment. It has been estimated that the Canadian content was approximately 70 % and supplied by nearly 600 private companies [AECL 1997]. First criticality was finally achieved on November 15, 1966. First generation of electricity came in January of 1967, and full commercial operation was declared on September 26, 1968. The two-year lag from initial criticality to commercial operations was due to required modification to coolant pumps and valves from excessive heavy-water leakage; further, space-limitations extended maintenance times [Whitlock 2005]. Douglas Point operated commercially at 220 MWe for 16 years and gradually improved in performance over time until shutdown in 1984 [Whitlock 2005].

3.5.4 Pickering A Units 1 and 2 – Initial Commercial Reactor

Pickering A (Units 1 and 2) was the first commercial scale HWR project. The design represented a scale-up by more than a factor of two, based on successful demonstration operations at the Douglas Point reactor facility. Portfolio planning studies performed by Ontario Hydro in 1964 estimated that reactors on the scale of 500 MWe would be needed to compete with the upcoming generation of coal-fired stations and to meet anticipated energy growth scenarios. Further, it was decided to couple these reactors in two-reactor units [AECL 1997]. Ontario Hydro thus decided to commit to two 500 MWe units and negotiated an agreement with the AECL in 1964. In 1967, Ontario Hydro committed to two additional unit - a total of four reactors at the Pickering site [AECL 1997].

Ontario Hydro would build, own, and operate the units, and AECL would do the nuclear design and act as engineering consultant. Ontario Hydro would invest the equivalent of coal-fired units, about 40% of the cost, and the Ontario and federal governments would provide 28% and 32% of the cost, respectively [AECL 1997]. However, since the operating cost of Pickering A was expected to be significantly less than that of a coal station, Ontario Hydro agreed to pay back the government investments from the resulting savings. Over time, the government investments were fully repaid and would form an important revenue stream for AECL [Whitlock 2016a]. The total cost of the first two Pickering A reactors (to both the federal government and Ontario Hydro) was reported at $393 to $420 million. Ontario Hydro has reported that the estimate for all four reactors in 1965 was $508 million, and that the total cost for all four Pickering A units was $716 million [Martin 2004, Winfield 2006].

The Pickering A reactors were a demonstrated success in their first decade of operation and operated at 515 MWe for each of the four reactor units. They demonstrated consistently high capacity factors (with exception of approximately 10% of the tubes in two units were replaced). Units 1 and 4 of Pickering A are still operating while units 2 and 3 were shut down in 2005 and are now in safe storage status. The cost of electricity produced by the Pickering A units was less than from contemporary coal-fired stations, which enabled the repayment to the provincial and federal governments mentioned above. Heavy-water leakage and loss were well within the design targets, and radiation fields remained low. In short, few of the early problems experienced at NPD and Douglas Point were encountered [AECL 1997]. The following three decades from 1959 to 1987 brought about twenty large commercial CANDU reactors that contributed about 15% of the electricity generated for Canada, overall, and more than 50% of the electric power generated in Ontario [AECL 1997, WNA 2015]. A further twelve CANDU's have been sold to international clients: South Korea (4 reactors), Romania (2), India (2), Pakistan (1), Argentina (1), and China (2)—as well as 13 CANDU-derivative reactors built by India [WNA 2015]. The extent of the government and industry roles in commercializing the HWR are presented in Table 3-3.

Table 3-3
Government and Industry Roles in HWR RD&D in Canada.

Reactor, [Dates of Operation], Location, Electricity Generation Capacity	**Government Role**	**Industry Role**
HWR (Canada) National Research Experimental reactor (NRX) Chalk River, Ontario [1947-1952; 1954-1992*] 10-40MWth	• AECL o Continued to perform the necessary R&D o Paid for most of the cost of the nuclear steam generating system.	• Ontario Hydro: o provided the site, o designed the balance of plant, o paid for the balance of plant, o Commissioned and Operated the station. o Reimbursed AECL for all energy produced at a competitive rate based on current market prices from coal generation • Canadian General Electric Company-Civilian Atomic Power Department (CGE-CAPD): o Designed the nuclear steam generating system, o Oversaw the plant construction, o Undertook manufacturing of certain components. • Contracts were solicited from private industries for the design and supply of most of the plant components
HWR (Canada) National Research Universal reactor (NRU) Chalk River, Ontario [1957-2018*] 135 MWth	• AECL: o R&D of reactor and fuel development o Paid for and owned the nuclear reactor	• None identified in the literature, , but most likely similar to that of the NRX excluding power production/balance of plant requirements

Table 3-3 (continued)
Government and Industry Roles in HWR RD&D in Canada.

Reactor, [Dates of Operation], Location, Electricity Generation Capacity	Government Role	Industry Role
HWR (Canada) Nuclear Power Demonstration-2 (NPD-2), Rolphton, Ontario, [1962-1985*] 20 MWe	• AECL: o R&D Support on the reactor and fuel development o Paid for and owned the nuclear reactor o Co-developed fuel for the CGE	• Ontario Hydro-led Nuclear Power Group (NPG)** Consortium:: o Supplied the site and the conventional power equipment, Purchased steam from the nuclear system and operate the plant o R&D Support on the reactor and fuel development o Paid for and owned the nuclear reactor o Operators and key staff members recruited and trained by Ontario Hydro • Canadian General Electric (CGE): o Co-developer of fuel with AECL o Engineering contractor • Canadian Bechtel Limited: o Construction of the plant under CGE
HWR (Canada) Douglas Point Kincardine, Ontario [1968-1984] 200 MWe	• AECL: o Designed and owned the plant o R&D support on the reactor and fuel	• Ontario Hydro: o Assisted with initial design study o Customer of the electricity o Plant constructor, operator • Canadian General Electric (CGE): o Engineering contractor
HWR (Canada) Pickering A, 4 stations, Pickering, Ontario, [1971-present day*] 515 MWe each, Over 1600 MWe combined capacity	• AECL: o R&D support on the reactor and fuel • Federal & Provincial Governments: Cost sharing of 40% of capital costs	• Ontario Hydro: o Builder, Owner, Operator o Customer of the electricity o Site owner, constructor, operator o Also returned savings back to Federal and Provincial Governments

3.6 United Kingdom’s RD&D to Commercialize MAGNOX

3.6.1 Research Reactors at Harwell – Test Reactors

The country's first nuclear reactors were built at the Atomic Energy Research Establishment (AERE) at Harwell in Oxfordshire. The site had been chosen at the beginning of 1946 by UK's Ministry of Supply, which had been assigned responsibility for nuclear research by the UK’s Atomic Energy Act [Sneddon 2010]. The first reactor at AERE was the Graphite Low Energy Experimental Pile (GLEEP); it was fueled with metallic natural uranium (clad in aluminum) and operated at 100 kWth. GLEEP had an air-cooled, horizontal configuration, and was graphite-moderated; it commenced operation in August 1947 and was operated for 43 years (prior to shut down in 1990)—during this time it functioned as a test reactor for the accumulation of nuclear data, such as neutron absorption properties of materials, and was also used for early production of radioisotopes [Hill 2013]. GLEEP was followed by the commissioning in 1948 of a larger test reactor, the 6 MWth British Experimental Pile '0' (BEPO). BEPO, like GLEEP, also had an air-cooled, horizontal configuration, was fueled with aluminum-clad natural uranium metal, and was graphite-moderated. These two test reactors at AERE provided technical and design information, along with operational experience to support the design and construction of the Windscale reactors (or "piles" as they are still known).

3.6.2 Windscale Production Reactors – Test Reactors

During 1947, the site of the former Sellafield ordnance factory – renamed Windscale – was announced as a new atomic energy site, and construction activities on the two Windscale reactors commenced. Windscale 1 and 2 were operated at approximately 30 MWth and fueled with aluminum-clad, metallic natural uranium; they were air-cooled and graphite moderated and were completed in 1950 and 1951, respectively [Hill 2013]. They produced plutonium (separated in a co-located chemical plant) but no electric power—the heated air was filtered prior to being vented to the atmosphere through 400-foot tall stacks. The reactors were shut down after a graphite fire [Wakeford 2007] in Unit 1 in 1957; the UK Atomic Energy Authority (AEA, which had taken over management of the atomic energy establishment in the UK in 1954 [Sneddon 2010]) evaluated restarting Unit 2, but the plutonium production capability of the reactors at Calder Hall and Chapelcross (below) was deemed to be sufficient [Hill 2013].

3.6.3 Calder Hall and Chapelcross Dual Use Reactors – Small Demonstration Reactors

In 1953, following the government announcement that the country would begin a civil nuclear power program, construction of the first nuclear power reactors at Calder Hall on the Windscale site—intended to supply both plutonium for the weapons program and electrical power—commenced. The reactors at Calder Hall were graphite-moderated and fueled with natural uranium metal; however, the design had been revised to a vertical configuration. They were cooled with CO2, and the metallic fuel was clad with a magnesium alloy known as MAGNOX (for magnesium alloy, non-oxidizing).

The design used at Calder Hall was given the designation PIPPA (pressurized pile producing industrial power and plutonium) [Hill 2013]. In the follow-on effort (below), when plutonium production was no longer a goal, the plants would be renamed MAGNOX. The PIPPA reactor design was developed and provided by the UKAEA, but commercial vendors would provide design and engineering services and construct the reactors and power generation equipment. A similar arrangement was used by the UKAEA to build a virtually identical plant at Chapelcross in Scotland. Both Calder Hall and Chapelcross were comprised of four reactors, each rated at approximately 60 MWe. All eight of these reactors would go into commercial power production between 1956 and 1960, and since both of these stations were required for plutonium production, their operation would remain with the UKAEA until taken out of service in 2003 [Taylor 2016].

3.6.4 Berkely, Bradwell, and Hunterston – Large Demonstration Reactors

Three consortia were set up in 1955 to conduct design and construction of large demonstration reactors based on the Calder Hall experience. Contracts were awarded shortly thereafter to: (1) Associated Electrical Industries (AEI), a consortium led by Thompson, for the contract at Berkeley; (2) Nuclear Power Plant Company (NPPC), a consortium led by C.A. Parsons (who had participated as a commercial contractor at Calder Hall and Chapelcross) for the contract at Bradwell; and (3) a consortium led by General Electric Company (GEC – not related to the U.S. company of the same name) for the Hunterston contract. These contracts involved 6 reactors, two at each site. Unlike the management arrangement for Calder Hall and Chapelcross, the design and engineering corsortia from this point forward were given broad leeway to revise the nuclear plant design, even down to the fuel element design.

3.6.5 MAGNOX Design and Deployment – Commercial Reactors

To support moving on to fully commercial development of nuclear power, a fourth and a fifth consortium was formed. The fourth consortium, set up in 1955, was led by English Electric for the contract at Hinkley Point. The fifth one, Atomic Power Constructions (APC), was led by International Combustion and was awarded the contract for Trawsfynydd. Trawsfynydd . These contracts involved four larger scale MAGNOX reactors, two per site, served to launch the commercialization of nuclear power in the UK. The broad leeway given to the designers of the large demonstration and commercial reactors resulted in variation in fuel element design and power ratings at each of the stations designed and constructed—indeed, it has been stated that "each site is unique" [Hill 2013]. Three additional UK MAGNOX sites would be developed, Sizewell A, Oldbury and Wylfa power stations, between 1960 and 1963 (again, two reactors at each site). The Wylfa stations were more than twice as large as prior MAGNOX units at 590 MWe per unit; however, they also experienced the longest construction time—98 months (about 40% longer than the average MAGNOX construction period). Also, single MAGNOX reactors would be sold to Italy and Japan, they were built at: Latina in Italia, brought on-line in 1963; and Tokai Moru in Japan, operational in 1965. All the UK MAGNOX reactors have now been decommissioned—Wylfa was the last to come off line in 2015 [Sneddon 2010]. The MAGNOX fleet provided an average of 40 years of commercial operation in the UK.

3.6.6 AGR, Next-Generation Nuclear Fleet – Commercial Reactors

The UK's MAGNOX fleet has been replaced with a next-generation, higher temperature gas-cooled reactor, the Advanced Gas-cooled Reactor (AGR, 640ºC versus about 350ºC for the MAGNOX design). The AGR went through a similar development sequence as the MAGNOX design, involving two purpose-built test reactors and a demonstration reactor, the High Temperature Reactor (HTR) built on the Windscale site in Sellafield, Cumbria. Highlights of the AGR design include: uranium oxide fuel (vice uranium metal for the MAGNOX design); stainless steel cladding, versus the magnesium alloy that gave the MAGNOX reactors their name; continued use of pre-stressed concrete pressure vessels (which had been incorporated into the latter MAGNOX designs), and a higher power rating—660MWe per reactor, which was about triple the average MAGNOX reactor rating (other than Wylfa) [RNL 1996]. The decision to build the AGR was conducted in an era of uncertainty, both as to what kind of reactors to build (LWRs and a UK-designed HWR were in the running) and whether to continue with nuclear power construction in the aftermath of the Windscale fire and, later, Three Mile Island. Design and development work on the AGR started in 1957 and the HTR would start operations in 1962; construction and design issues, coupled with extended programmatic uncertainty, led to the first AGRs not entering commercial operation until 1985 (construction started in 1965). Subsequent stations would average about a 10-year construction period and there are presently 14 AGRs operating in the UK.

Table 3-4, presents the four projects that made significant contributions to the commercialization of the UK's gas-cooled reactor, MAGNOX [Sneddon 2010, Hill 2013, RNL 1998].

Table 3-4
Government and Industry Roles in GCR RD&D in UK.

Reactor, [Dates of Operation], Location, Electricity Generation Capacity	Government Role	Industry Role
GCR (UK) Graphite Low Energy Experimental Pile (GLEEP) [1947-1990] 100 kWth; Also the British Experimental Pile '0' (BEPO) [1948-1968] 6 MWth	• AERE, Ministry of Supply & Ministry of Works): ○ R&D: Pre-Construction ○ Designer ○ Owner ○ R&D: Post-Construction ○ Fuel Fabricator & Supplier	• None Identified
GCR (UK) Windscale 1 & 2 [1952 – 1957], 30 MWth Sellafield, Cumbria	• AERE [then UKAEA], Ministry of Supply & Ministry of Works): ○ R&D: Pre-Construction ○ Designer ○ Owner ○ R&D: Post-Construction ○ Fuel Fabricator & Supplier	• None specifically identified; reference, however, made to "suppliers and contractors" under the Ministry of Works
GCR (UK) Calder Hall [1956-2003] 60 MWe in each of 4 reactors (240 MWe total), Sellafield, Cumbria	• AERE [then UKAEA], Ministry of Supply & Ministry of Works: ○ R&D: Pre-Construction ○ Designer ○ Owner ○ R&D: Post-Construction ○ Fuel Fabricator & Supplier • British Electrical Authority (BEA), then Central Electricity Generating Board (CEGB): ○ Operated the steam supply system and power plant	• Parolle Electrical Plant Company of Newcastle upon Tyne: ○ Independent engineering review of the design • Private companies involved as A/E firm, and as construction contractors
GCR (UK) Berkeley 1 [1962-1989] 138 MWe, Berkeley, Gloucestershire	• UKAEA: ○ R&D: Pre-Construction ○ Nuclear Plant Design Advice & Guidance ○ Continued R&D: Post-Construction ○ Fuel Fabricator & Supplier • CEA: ○ Plant Owner & Operator	• Nuclear Energy Company (NEC) was led by Associated Electrical Industries Ltd and John Thompson Limited, along with 4 site-specific contractors: ○ Applied R&D ○ Fuel System Modifications ○ Reactor Plan Layout ○ A/E Services ○ Power plant design, construction management, and trained operators
GCR (UK) Hinkley Point – A [1965, 250 MWe]	None Identified	• English Electric

3.7 US Non-LWR Development Efforts

Three research reactors were the basis of proving the basic technical features of the HWR: Zero Energy Experimental Pile (ZEEP), National Research Experimental (NRX) reactor, and National Research Universal (NRU) reactor. A detailed proposal for ZEEP was developed in 1944 as a part of the Manhattan Project. ZEEP was designed to operate at 1 watt (thermal); it was fueled with natural uranium and used heavy water as a moderator. The first criticality was reached on September 5, 1945 [AECL 1997].

The original proposed reactor development program by the AEC outlined to Congress in 1954 called for an R&D program for five reactor technologies. Relative merits of the proposed projects were divided into categories of developmental time frames by the AEC [JCAE 1954]:

- Short term: ready for large experimental testing in 2 or 3 years with a good chance of mechanical success (included the PWR and sodium graphite reactor (SGR))
- Middle term: ready for testing on a large scale within 5 years (included the BWR)
- Long term: ready for large experimental testing in no less than 5 years unless some unexpected technical break-through occurred (included the homogeneous reactor and fast breeder reactor)

The AEC also evaluated the proposed projects for their long-term potential to produce low-cost electric power. The consensus of opinion for achievement of economically competitive nuclear power, as expressed by the AEC, was nearly inverse of the list of the estimated developmental timeframes for each of the five originally proposed reactor concepts, as shown in Table 3-5.

Table 3-5
Level of Promise and Estimated Required Developmental Timeframes for AEC's Early Reactor R&D Program.

Level of Promise (1 is most promising; 5 is least promising)	**Developmental Timeframes** (1 is shortest; 5 is longest)
1. Homogeneous reactor	1. Pressurized water reactor
2. Fast breeder reactor	2. Sodium graphite reactor
3. Boiling water reactor	3. Boiling water reactor
4. Sodium graphite reactor	4. Homogenous reactor
5. Pressurized water reactor	5. Fast breeder reactor

Although the pursuit of the non-LWR concepts discussed below did not result in commercialized systems, the effort to develop these systems is relevant to the discussion of RD&D planning. It is important to understand where work had ceased and what RD&D efforts remain to forge a pathway to commercialization for these concepts, while also describing challenges that ultimately led to a project's termination. The following sections review the US government and industry efforts to develop the sodium-cooled reactor (both thermal and fast spectrum), high-temperature gas-cooled reactors (HTGRs), the homogeneous reactor (which includes the molten salt reactor, MSR), and the heavy water reactor (HWR).

3.8 Sodium-Cooled Reactor (Fast and Thermal Spectrum)

Sodium-cooled reactors – both thermal and fast spectrum – were postulated in the early 1950s to hold promise of inherently safer reactor behavior and efficient fuel utilization. In particular, the fast spectrum sodium-cooled reactors were pursued for their promise of breeding fuel, thereby potentially eliminating the need for new fuel [JCAE 1954]. Thermal spectrum sodium-cooled reactors were given less attention and thus the fast spectrum sodium-cooled reactors are discussed first and the thermal spectrum small demonstrations discussions follow.

The original proposed reactor development program by the AEC outlined to Congress in 1954 called for an R&D program for five reactor technologies. Relative merits of the proposed projects were divided into categories of developmental time frames by the AEC [JCAE 1954]:

- Fast-Spectrum Sodium-Cooled Reactors:
  - Los Alamos Molten Plutonium Reactor Experiment (LAMPRE-I) at the Los Alamos National Laboratory (LANL), NM
  - Experimental Breeder Reactor I and II (EBR-I and EBR-II) at Arco, ID
  - Enrico Fermi in Monroe, MI
  - Southwest Experimental Fast Oxide Reactor (SEFOR) near Fayetteville, AR
  - Fast Flux Test Reactor (FFTF) at the Hanford site in Richland, WA
  - Clinch River Breeder Reactor (CRBR) in Oak Ridge, TN
  - Large Scale Prototype Breeder (LSPB) in Naperville, IL
- Thermal-Spectrum Sodium-Cooled Reactors:
  - Sodium Reactor Experiment (SRE) in Santa Susana, CA
  - Hallam Sodium Graphite Reactor (SGR) in Columbus, NE

3.8.1 Los Alamos Molten Pu Reactor Experiment (LAMPRE) – Test Reactor

Los Alamos National Laboratory (LANL) developed and briefly operated the Los Alamos Molten Plutonium Reactor Experiment (LAMPRE) which was a sodium-cooled reactor fueled with molten plutonium . It achieved initial criticality in early 1961 and operated successfully for several thousand hours until mid-1963. LAMPRE was designed to explore issues associated with using plutonium fuel in fast breeder reactors and was originally intended to operate at 20 MWth. It became apparent, however, that knowledge was inadequate about the behavior of some of the core materials in a high-temperature, high-radiation environment. The design power therefore was reduced to 1 MWth, with the plan to follow LAMPRE-I by a 20 MWth LAMPRE-II. By mid-1963, LAMPRE-I had served its intended purpose and was shut down. Financial support for the LAMPRE-II project was not approved and did not move forward [Cochran 2009, Bunker 1983, LANL 2004].

3.8.2 Experimental Breeder Reactor I and II (EBR-I and EBR-II) – Test Reactor and Small Demonstration Reactor

The Experimental Breeder Reactor I (EBR-I) made history on December 20, 1951 when it supplied sufficient power to light four light bulbs at the NRTS site in Arco, Idaho. The achievement was significant because it demonstrated the technical feasibility of electricity generation from nuclear energy [DOE 2011]. Although important, EBR-I's real mission was not to show that electricity could be generated by a nuclear reactor; it was to determine that fuel breeding could actually be achieved [INL 2012]. EBR-I was rated at 1.4 MWth (150 kWe) and used sodium-potassium (NaK) coolant with plutonium fuel [AEC 1972].

EBR-I was designed and constructed by ANL with construction beginning in 1949 at a cost of $2.73 million in 1957 dollars [JCAE 1957a]. During EBR-I's operational lifetime, multiple EBR-I cores were developed in which the last of them, Mark IV, produced 1.27 new atoms of fuel for each atom consumed. An accident in EBR-I led to 40-50% of the core melting on November 29, 1955 [Haroldsen 2008, ANL 1958, Kittel 1958]; however, the reactor was rebuilt and continued to be used for research until December 1964 [AEC 1972].

EBR-I's successor, the EBR-II, operated from 1964 to 1994 at the NRTS site in Idaho [INL 2012]. Preliminary planning and feasibility studies conducted by the AEC for EBR-II started in December 1953, and funding approval for almost $15 million occurred in July 1955 [Westfall 2004]. Later in 1957, the costs of EBR-II were estimated to be approximately $15.3 million [JCAE 1957a]. EBR-II was a sodium-cooled reactor using a mix of plutonium and uranium oxide fuel with a heat output of 62.5-MWth and electrical capacity of 16.5 MWe [AEC 1972, Walters 1999]. EBR-II achieved a capacity factor of 70% while completing several research campaigns regarding nuclear materials in fast-spectrum environments [Walters 1999]. On April 3, 1986 two milestone tests were conducted in EBR-II. The first test was a loss of flow without scram, and the second was a loss of heat sink without scram. Both tests were initiated from 100% power, and in both tests the reactor was shut down by inherent physical processes, principally thermal expansion, without automatic scram, operator intervention, or the help of special in-core devices. The temperature transients during the tests were mild, as predicted, and there was no damage to the core or reactor plant structures [Walters 1999, Planchon 1987].

The EBR-II reactor was also intended to serve as the prototype for the Integral Fast Reactor (IFR), a reactor fueled by metal alloy and cooled by liquid sodium developed at Argonne National Laboratory in the decade 1984 to 1994. The IFR project was to develop the technology for a complete system; the reactor, the entire fuel cycle, and the waste management technologies were all to be included in the development program [JCAE 1955, JCAE 1952]. Eventually, the IFR project was ended and not constructed [ANL 2016c]. The EBR-II tested many of the components for the Fast Flux Test Facility (FFTF) which was constructed in Richland, Washington and the Clinch River Breeder Reactor (CRBR) project that was not constructed (as described in the following sections) [ANL 10991].

3.8.3 The Fermi Reactor - Small Demonstration Reactor [PDRP]

The Enrico Fermi reactor, awarded in March 1957, was a joint effort pursued by the AEC and the Power Reactor Development Company (PRDC) during the first round of the PDRP [JCAE 1963]. The fast breeder reactor was located at Monroe, Michigan and was designed to reach 400 MWth but achieved heat output levels of around 200 MWth and 66 MWe during the first core operations [NRC 2016f]. Sodium was used as the coolant and plutonium oxide and uranium oxide as fuel [AEC 1972].

The PRDC of Detroit, Michigan was a corporation owned by 23 utility and industrial companies. In the 1957 agreement with the AEC, PRDC agreed, at its own expense, to design, develop, construct, own, and operate the nuclear power plant. PRDC also agreed to arrange for the Detroit Edison Company to build, own, and operate a generator plant with a maximum capability of 150 MWe to use the steam produced by the reactor. PRDC contracted with Atomic Power Development Associates (APDA) to furnish various development, design, engineering, and fabricating services required for the construction and operation of the reactor. AEC agreed to provide the $4.5 million of R&D work in AEC-owned laboratories, in addition to consultant and information services, and assistance in training reactor plant operators and maintenance personnel. The AEC also agreed to waive fuel use charges for the first 5 years of operation (estimated at about $5 million) [JCAE 1963]. The total construction costs were originally estimated at $47 million, of which about $33 million would be for the reactor and for equipment associated with the reactor portion of the plant.

The Fermi project encountered licensing problems during the construction permit hearings [NSF 1977b] and delayed the start of construction to December 1956. By 1966, construction costs had risen to $51 million, and the total costs of the project were estimated to be over $102 million [JCAE 1966]. Criticality was first reached in August 1963, and grid connection occurred in August 1966. In 1965, the AEC negotiated a memorandum of understanding with the PRDC on the provisions of a proposed contract for irradiation services in support of the AEC's fast reactor fuel development program continuing through May 1968. PRDC subcontracted important elements of the work to APDA with AEC approval. PRDC was obliged to provide advice and assistance to AEC and its contractors in designing the AEC fuel and material to be irradiated in the Fermi reactor, and the APDA would be reimbursed for costs incurred directly in connection with the AEC work. PRDC furnished, without charge to the AEC, the design of the standard test subassembly for irradiation experiments in the reactor, as prepared by APDA. The operation of the plant was the exclusive responsibility of PRDC [JCAE 1966]. The PRDC base plan of operation followed a step-by-step increase in power to 200 MWth [JCAE 1966].

In October 1966, Fermi suffered a partial core meltdown during a power ascension [NRC 2016f]. The reactor was repaired by 1970 and continued operations until 1972 when the PRDC decided to decommission the plant. It has been observed that Fermi had experienced operational challenges attributed to technical immaturity [NSF 1977b].

3.8.4 Southwest Experimental Fast Oxide Reactor (SEFOR) - Test Reactor

The Southwest Experimental Fast Oxide Reactor (SEFOR) was designed and constructed as a 20 MWth, sodium-cooled transient test reactor. It was designed for power oscillations, sub-prompt critical excursions, and super-prompt critical excursions to demonstrate the safety characteristics of oxide fuel [INL 2010, BNL 1994]. The AEC contracted with General Electric to design, construct, and operate the reactor near Fayetteville, Arkansas. Nuclear Fuel Services (NFS) fabricated the fuel rods from AEC-supplied plutonium from the Hanford site, operated by the Atlantic Richfield Hanford Company (ARHCO). While General Electric was principally responsible for designing, constructing, and operating the reactor, other project collaborators were the Kernforschungszentrum Karlsruhe (KFK), Euratom, and Southwest Atomic Energy Associates, a consortium of 17 utilities [BNL 1994].

Construction of the SEFOR project lasted between 1965 and 1968. Mixed oxide fuel of plutonium and uranium oxide was used [INL 2010]. Initial operations were delayed due to issues related to the plutonium content in the received fuel [BNL 1994]. SEFOR began operating in May 1969, and was shut down three years later. The fuel and irradiated sodium coolant were removed and taken offsite in 1972, and some dismantling was performed. The reactor was acquired by the University of Arkansas in 1975 and is still owned by the university, although the university has never operated it [Cochran 2009].

3.8.5 Fast Flux Test Facility (FFTF) – Test Reactor

The AEC placed high priority on the development of the Liquid Metal Fast Breeder Reactor (LMFBR) in the late 1960s [DOE 1983]. Goals of demonstrating the LMFBR technology by 1980 were outlined in 1971 that included two projects: a test reactor in Richland, Washington – the Fast Flux Test Facility (FFTF) and a demonstration plant in Oak Ridge, Tennessee—the Clinch River Breeder Reactor (CRBR, as discussed in the next section). The FFTF was a 400 MWth sodium-cooled fast reactor completed in 1978. Initial criticality was achieved in early 1980, and full power achieved on December 21, 1980 [DOE 1983, DOE 2015a]. The facility was constructed to support development and testing of fuels, materials, and equipment under prototypic conditions for the Liquid Metal Fast Breeder Reactor program. [Nielsen 2004].

FFTF was an example of a DOE-regulated facility that used the NRC to perform the technical review. Since the NRC was formed in 1974 after the initial design of the FFTF had been completed, the Preliminary Safety Analysis report, available in 1970, had been submitted to the AEC. Nonetheless, the AEC/DOE requested that the NRC and the Advisory Committee on Reactor Safety (ACRS) perform a technical review to "bring the NRC up to speed on licensing sodium-cooled fast reactors" [DOE 2015b]. The NRC issued a Safety Evaluation Report in August 1978 prior to the start of FFTF's operation [DOE 1978, NRC 1979]. During the construction of the FFTF, actual costs increased over original estimates, and the schedule to completion was drawn out. In late 1993, DOE decided not to continue operating FFTF. In 2009, FFTF completed deactivation activities and was placed in a long-term, low-cost surveillance and maintenance condition [DOE 2015a, Farabee 2006].

3.8.6 Clinch River Breeder Reactor (CRBR) – Large Scale Demo (intended)

The CRBR was to be a 375 MWe (975 MWth) fast breeder reactor, building upon the FFTF experience, and it promised to be a next step in the transition to a large-scale demonstration of the fast breeder concept [DOE 1982a]. The CRBR project was launched in August 1972 with the signing of a memorandum of understanding between the AEC and the principal utility participants, the Commonwealth Edison Company and the Tennessee Valley Authority (TVA). The AEC would be responsible for R&D of the demonstration plant while the Commonwealth Edison Company and TVA would engineer, manufacture, and proof test equipment and systems [DOE 1983]. The CRBR was to be licensed as a commercial reactor under the NRC purview with licensing work beginning in 1974. The Carter administration attempted to halt the licensing process but the work continued in 1981. In 1983, Congress stopped the licensing process; however, the CRBR did receive a limited work authorization that same year. The NRC did not identify technical issues of concern during its review, but no construction permit was granted; support for the project in Congress waned and the project was discontinued [DOE 2015b]. In a 1981 report, it was estimated that $1 billion had already been spent on Clinch River compared with an original cost estimate of $700 million [Boudreau 1981].

3.8.7 Large Scale Prototype Breeder (LSPB) - Large Demonstration Reactor (intended)

The Large Scale Prototype Breeder (LSPB) was a cooperative effort between the DOE and the Electric Power Research Institute (EPRI) with the intention to include financial support from electric utilities, private industry, and foreign participants. The EPRI Consolidated Management Office (EPRI-Como) in Chicago directed the program to design the LSPB [DOE 1987]. Como was established in June 1982 in Naperville, Illinois and was responsible for planning, implementing, and managing the technical program of EPRI's new Advanced Nuclear Generation Department whose mission was to identify and develop advanced breeder and improved light water reactor systems [EPRI 1984].

It was anticipated that private industry, led by EPRI, would take the prime responsibility for constructing and operating the LSBP [DOE 1982b]. Participating contractors in the LSPB design included General Electric, Rockwell International, Combustion Engineering, Westinghouse Electric, Bechtel, Burns and Roe, and Stone and Webster [DOE 1987]. An agreement was signed with Japan's Federation of Electric Power Companies to expand international collaboration efforts and to study other advanced nuclear plant designs, including designs for pool-type breeders [EPRI 1984].

The LSPB design features included the pool concept, utilizing a mixed oxide fuel and a four-loop configuration plant at a capacity of 1370 MWe reactor (3500 MWth) [DOE 1987]. Advantages over previous breeder designs were postulated, such as passive decay-heat removal, elimination of guard vessels around primary components, and out-of-vessel fuel storage tank designed to be housed within the containment building. In an EPRI 1984 report [EPRI 1984], another beneficial aspect of the LSPB design was that “in contrast with earlier designs, the reactor vessel head remains in place during refueling and the containment above the head is in an air atmosphere, making it accessible to personnel during plant operation” [EPRI 1984]. The project ended in 1986 and the final documentation developed for the LSPB concept included: (1) design description report; (2) plant system/building design/analysis reports; (3) plant-wide assessments; (4) set of generic safety reports for NRC review; (5) cost estimate report; and (6) status evaluation report [DOE 1987].

3.8.8 Sodium Reactor Experiment (SRE) – Test Reactor

The AEC and Atomics International, a division of North American Aviation (NAA) Incorporated, jointly funded the Sodium Reactor Experiment (SRE) located at Santa Susana, California, as part of AEC’s five-year reactor study before the start of the PDRP [JCAE 1954]. The SRE was designed to resemble a full-scale plant and included a pool-type reactor [JCAE 1954]. Two other objectives of the SRE were to provide a firmer basis for estimating the performance and economics of this reactor technology and to test and develop the advanced technology necessary to attain economically competitive power for subsequent plants [JCAE 1956].

The contract between the AEC and NAA in 1955 outlined expenditures of $10 million from start of the project through 1958. Approximately 40% was allocated for fabrication of the reactor and associated buildings, and the remaining 60% was to be used for R&D. NAA was to provide $2.5 million of the funds required. During a 1957 update, the cost of the reactor had increased to $6.85 million and R&D costs through fiscal year 1956 were $7.4 million [JCAE 1957a].

Original plans did not provide for any generation of steam or electricity in 1955; however, AEC entered into contract with the Southern California Edison Company in which the utility would provide the necessary boiler and turbogenerator rated at 7.5 MWe to add onto the 20 MWth reactor [JCAE 1955, JCAE 1956]. NAA would design, fabricate, and operate the facility located at the Santa Susana Field Laboratory, owned by Boeing and NAA [JCAE 1956, Ragheb 2015].

Initial criticality of the SRE was reached on April 25, 1957, attaining a power level of 5.8 MWe and 20 MWth [Ragheb 2015]. The SRE used a metal fuel mixture of uranium-235 and thorium-232 such that breeding of uranium-233 material would occur [JCAE 1954]. The heat produced could also be exhausted to the atmosphere through a sodium-to-air heat exchanger [Ragheb 2015]. Grid-connected power production began on July 12, 1957 but was halted on July 29, 1959 due to several fuel elements failing from localized overheating, and fuel fragments had fallen to the bottom of the reactor. The cause had been traced to a failure in the impellor pump when the pump seal coolant seeped into the primary coolant system and decomposed into a hardened residue blocking the sodium cooling channels [Ragheb 2015]. It is noted in [Ragheb 2015] that "the reactor excursion is explained by the expulsion of sodium from several of the partially blocked fuel channels. Since sodium acts in this system as an absorber, its expulsion results in a positive reactivity insertion in a system that has a positive power coefficient of reactivity" [Ragheb 2015]. The SRE was returned to service on September 7, 1960 after the core was replaced and the primary coolant loop pumps were redesigned. The unit provided test information and power until 1966 [DOE 2011].

3.8.9 Hallam Sodium Graphite Reactor (SGR) – Small Demonstration Reactor [PDRP]

One of the proposals submitted to the AEC during the first round of the PDRP was by the Consumers Public Power District of Nebraska for the Hallam Sodium Graphite Reactor (SGR). The initial planning and design was carried out in 1949 by the Atomics International Division of North American Aviation (NAA) [Beeley N/A]. SGR's proposed capacity was 75 MWe (240 MWth) and was based on the SRE (above) [JCAE 1955, Beeley N/A]. The AEC relaxed Consumer's financial responsibilities for construction and operations due to its limited ability to assume such risks as a publicly owned organization. As a result, the utility agreed to provide only 25% of the estimated total costs in the final contract [JCAE 1963, 10]. Consumers' proposal was officially accepted under the first round of the PDRP, but has been described as closely approximating the terms of the second round due to its funding utility and the cost share percentage [NSF 1977b]. Also, since the AEC entered into contract with Consumers well into the second round of the PDRP, the AEC adapted some of the provisions to match new elements of the second round with Consumers. Consumers would contribute up to $5.2 million of the $24 million toward the cost of constructing the nuclear portion of the plant. The AEC would cover the rest of the reactor construction costs, the supporting R&D at a cost of $18 million, and would waive the first core fuel use charges, estimated at a value of $1.3 million. The total cost of the SGR was estimated to be approximately $57M in 1956 dollars [JCAE 1963].

Table 3-6
Summary of US Sodium-Cooled Reactors Development (Fast).

Plant	Location	Owner	Initial Criticality Date [Operating dates]	Size [MWe, MWth]	Type of Facility [Experiment/Test, Small-Scale Demonstration, Large-Scale Demonstration, Commercial]	Stage of development
Clementine* (Hg-cooled)	LANL site (NM)	Gov't (AEC)	Nov 1946 [1949-1950; Sept 1952-Dec 1952]	25 kWth	Experiment/Test	C (constructed, limited ops)
EBR-I	NRTS (ID)	Gov't (AEC)	August 1051 [1951-1955; 1956-1964]*	1.4MWth; 150kWe	Experiment/Test	C (constructed, limited ops)
EBR-II	NRTS (ID)	Gov't (AEC)	Nov. 1962 [1964-1994]*	62.5MWth 16.5MWe	Small Demo	C (constructed, limited ops)
Integral Fast Reactor (IFR)	NRTS (ID)	Gov't (AEC)	--	--	Small Demo	B (designed, not constructed)
LAMPRE-I	LANL site (NM)	Gov't (AEC)	Early 1961 [1961-1963]	1 MWth	Experiment/Test	C (constructed, limited ops)
LAMPRE-II	LANL site (NM)	Gov't (AEC)	--	20 MWth	Small Demo	B (designed, not constructed)
Fermi-I	Monroe, MI	Gov't (AEC)	August 1963 [Aug 1966-Dec 1966; 1970-1972]*	200MWth 66MWe	Large Demo	C (constructed, limited ops)
SEFOR	Fayetteville, AK	Gov't (AEC); Private (Univ. of AK)*	[1969-1972]*	20 MWth	Small Demo	C (constructed, limited ops)
FFTF	Richland, WA	Gov't (AEC)	Early 1980 [1980-1993]	400MWth	Large Demo	C (constructed, limited ops)
CRBR	Oak Ridge, TN	Common-wealth Edison and TVA	--	975MWth 375MWe	Large Demo	B (designed, not constructed)
LSPB	--	Private industry, led by EPRI and DOE	--	3500 MWth 1370 MWe	Initial Commercial	B (designed, not constructed)

The AEC would retain ownership of the nuclear reactor and Consumers would operate the entirety of the plant while also owning the electrical portion of the plant. The AEC contracted NAA and Bechtel for research, development, design, and other services on the nuclear plant. Peter Kiewit Sons, Incorporated, was brought on as the general construction contractor for the nuclear plant [JCAE 1963]. Construction began in April 1959 and was completed in October 1961. Operations of the SGR were on-again, off-again during 1962-1964 [DOE 1979]. During the time of SGR's operation, sodium was used as the coolant with low enriched uranium fuel and a graphite moderator. Even though the SGR concept had been considered promising a decade earlier, it experienced technical problems with heat exchanger failures, and the AEC halted the project in 1964 [NSF 1977b, Cochran 1964]. Hallam was dismantled by NAA between 1964 and 1968 [DOE 1979].

Table 3-7
Summary of US Sodium-Cooled Reactors Development (Thermal).

Plant	Location	Owner	Initial Criticality Date [Operating dates]	Size [MWe, MWth]	Type of Facility [Experiment/Test, Small-Scale Demonstration, Large-Scale Demonstration, Commercial]	Stage of development
SRE	Santa Susana, CA	North American Aviation & Southern California Edison	April 1957 [1957-1959; 1960-1966]*	20 MWth 7.5MWe	Small Demo	C (constructed, limited ops)
Hallam SGR	Hallam, NE	Gov't (AEC)	[1963-1964]*	240MWth 75MWe	Large Demo	C (constructed, limited ops)
Chugach Alaska	Anchorage, AK	Gov't (AEC)	--	40MWth 10MWe	Small Demo	A (proposal developed, no award)*

3.9 High Temperature Gas Reactor (HTGR)

The world's first reactor was the Chicago Pile (CP-1), which utilized graphite layers interspersed with uranium oxide and metal as fuel and was air-cooled. On December 2, 1942, the initial criticality of CP-1 marked a historically significant milestone for the harnessing of nuclear energy [ANL 2016b]. The earliest planning and design of gas-cooled, graphite-moderated reactors in the US was tied to the study of basic nuclear properties; gas reactors were also evaluated as an option for potential use for plutonium production at the Hanford site in Washington during the early 1940s. Use of water coolant instead of gas was decided upon for the Hanford reactors due to its greater cooling capacity for this first-of-a-kind, large-scaled production reactor [Hanford 2016, ORNL 1958]. Another early graphite-moderated and air-cooled reactor, the so called "Graphite Reactor," was constructed at ORNL and is discussed briefly below [ORNL 1958].

Although the U.S. constructed these two early gas-cooled reactors for defense purposes, earnest efforts by the U.S. to develop the technology for peacetime would lag behind the UK, which successfully commercialized the gas-cooled reactor in the 1950s and 1960s. As authorized by provisions within its 1959 authorization act, the AEC invited industry to participate in developing a multi-purpose test reactor that would be gas-cooled and graphite-moderated. However, what industry proposed was the Peach Bottom reactor near Harrisburg, Pennsylvania, which was designed to be a commercial power reactor without the flexibility of performing experiments and materials testing (as specified in the 1959 act). Despite this inconsistency, the AEC approved the Peach Bottom project (described below) and, in parallel, commissioned a project at ORNL to design and construct a gas-cooled test reactor [JCAE 1963]. The AEC cancelled the ORNL project in the late 1960s when the conceptual design produced by ORNL turned out to cost twice the amount of the AEC's original estimate. [ORNL 1958].

The inception of the high-temperature gas reactor (HTGR) for civilian peacetime use has been attributed to the General Atomic Division of General Dynamics Corporation which developed an HTGR concept in 1958 [NRC 2004]. The originally proposed advantages of HTGRs remain the same today – better thermal efficiencies for generating electricity and the potential to provide heat for high-temperature applications for industrial purposes (e.g., glass and cement manufacturing, chemicals production) [GIF 2002]. Two HTGRs that were constructed and operated within the US, along with the pioneer Graphite Reactor, are discussed below.

3.9.1 Graphite Reactor – Test Reactor

The Graphite Reactor, originally called "The Pile" at ORNL, was the proving grounds for plutonium production and separations technology which was later scaled to industrial production at the Hanford site in Washington. Within one year, the ORNL Graphite Reactor had been constructed and had produced sufficient quantities of plutonium for testing. The first criticality was reached on November 4, 1943, and the reactor was shut down 20 years later [ORNL 2009]. Aluminum clad uranium metal slugs were used within the air-cooled, graphite-moderated core. The power level was designed initially at 1 MWth, but was increased over time to 4 MWth [ORNL 2009]. The Graphite Reactor provided initial gram quantities of plutonium for verifying material and nuclear properties. It was used for materials research and for training until it was shut down in 1963. It is maintained by ORNL today as a national historic landmark.

3.9.2 Peach Bottom – Small Demonstration Reactor

During the modified third round of the PDRP in September 1958, the AEC invited proposals specifically to develop, design, construct, and operate a gas-cooled, graphite-moderated nuclear power plant of sufficient size to serve as an effective prototype for a future full-scale power plant of similar design [JCAE 1963, ORNL 2009]. A joint proposal for a project was submitted in November 1958 by the Philadelphia Electric Company and General Atomics to build a 40 MWe (115 MWth) nuclear power plant that would be located at Delta, Pennsylvania, near Harrisburg. The proposal was accepted by the AEC as a basis for contract negotiation in February 1959, and an agreement between Philadelphia Electric and the AEC was finalized in August 1959 [JCAE 1963]. Under the agreement, Philadelphia Electric would be the owner/operator and would assume responsibility for the design and construction of an advanced high-temperature, graphite-moderated, helium-cooled nuclear reactor, estimated to cost $27.2 million. The AEC waived the fuel use charges through the first five years of operation up to a maximum of $2 million, and, under a separate contract with General Atomics, agreed to conduct preconstruction and post-construction R&D valued at approximately $14.5 million [JCAE 1963]. The Bechtel Corporation would act as the architect-engineer [Everett 1978].

Peach Bottom's years of operation spanned from 1966 to 1974 [PNNL 2011]. During that time, two cores were cycled through, with the first core fueled with pyrolytic carbon coated thorium and uranium particles. The single coated pyrolytic carbon coated fuel particles released both solid particulate and gaseous fission products and thus, the second core was fueled with particles having two layers called bistructural isotropic (BISO) fuel. The BISO inner layer mitigated release of solid particulate fission products while the second layer retained the noble gases. The first core lasted until June 1970 when fuel failure was observed from fuel swelling and cracking of the cladding [Everett 1978, PNNL 2011]. Peach Bottom operated with the second core until the end of October 1974 and was shut down and slated for decommissioning. Decommissioning was completed in 1978 [Everett 1978]. A number of operational issues arose with the first core's graphite matrix fuel; a second issue involved primary coolant fouling with helium compressor lubricating oil. Peach Bottom was considered a successful small demonstration plant with a reported 37% thermal efficiency and an overall capacity factor of 75% [Everett 1978].

3.9.3 Fort Saint Vrain (FSV) – Large Demonstration Reactor

Concurrent with General Atomics' conceptual design development of the HTGR in the late 1950s, the Public Service Company (PSC) of Colorado was researching alternative power generation options outside of the PWR and BWR technologies [NRC 2004]. In March 1965, PSC and General Atomics announced plans to build the Fort Saint Vrain (FSV) facility north of Denver. On November 1st, PSC, General Atomics, and the AEC signed a three-party contract launching the effort to build a 330 MWe (842 MWth) nuclear power plant that was a helium-cooled, graphite-moderated reactor utilizing uranium-235 and thorium-232, as an initial effort to implement a thorium-uranium-233 fuel cycle [Habush 1974]. Construction began in September 1968. General Atomics was the prime contractor responsible for the design, procurement of equipment and materials, and construction of the nuclear steam supply system and the turbine plant. PSC was responsible for supplying and preparing the site. Sargent & Lundy subcontracted to General Atomics as the architect-engineering firm, and EBASCO was the principal constructor [Habush 1974, ORNL 1976]. The construction cost at completion was estimated to be approximately $200 million [Johnson 2014].

Table 3-8
Summary of US High-Temperature Gas Reactor Development.

Plant	Location	Owner	Initial Criticality Date [Operating dates]	Size [MWe, MWth]	Type of Facility [Experiment/Test, Small-Scale Demonstration, Large-Scale Demonstration, Commercial]	Stage of development
Chicago Pile (CP-1)	Chicago, IL	Gov't (AEC)	December 2, 1942 [1942-1943]	0.5 watts thermal	Experiment/Test	C (constructed, limited ops)
Graphite Reactor	ORNL site	Gov't (AEC)	November 1943 [1943-1963]	1-4MWth	Experiment/Test	C (constructed, limited ops)
Peach Bottom	Delta, PA	Gov't (AEC)	[1966-1974]	40 MWe	Small Demo	C (constructed, limited ops)
Fort Saint Vrain	Denver, CO	Public Service Company (PSC) of Colorado	January 1974 [1976-1989]	330MWe	Large Demo	C (constructed, limited ops)

During the 10 years it was operational, FSV produced an average of only 14.6% of its capacity due in part to a range of technical issues including water and gas incursion, fuel failures or anomalies, and failures and cracks in certain reactor components. [ORNL 2004]. In August 1989 during an unplanned maintenance outage to repair a stuck control rod pair, cracks were discovered in several steam inlet ring headers in the steam generator. The required repairs were determined too costly by PSC and led to the shutdown of FSV in August of 1989 [NRC 2004].

3.10 Homogenous Reactors & Molten Salt Reactors

The AEC was interested in developing the homogenous reactor as part of its initial original proposed reactor development program, but acknowledged that the technical maturity at the time was lagging behind the PWR and BWR [JCAE 1954]. The homogeneous reactor concept eventually evolved to include what is commonly referred to as the molten salt reactor (MSR), which represented the shift from an aqueous solution containing either uranyl nitrate or sulfate as the working fuel fluid to fissile material dissolved within a molten salt [ORNL 2009]. Initial work was led by the Los Alamos National Laboratory (LANL) which carried out tasks for defense purposes to design, construct, and operate a number of homogeneous reactors as small experiments to calculate critical masses of homogeneous solutions of U-235. Peacetime research was led principally by ORNL to research and develop both the homogenous reactor and the MSR [ORNL 2009]. Four experimental reactors of this type were designed, built, and operated by ORNL and are described within this section.

Other efforts to develop homogeneous reactors in the US were led by private industry. During the second and third rounds of the PRDP, the AEC received two design proposals from the Wolverine Electric Company and the Pennsylvania Power & Light (PP&L) Company, respectively. Both proposed a heavy-water moderated, homogeneous reactor design with Wolverine Electric offering a smaller scale design of 10 MWe (38 MWth) and PP&L putting forth a design ranging from 70 to150 MWe; however, neither one progressed past the initial planning and design stages [IAEA 2002, Morris 1992]. Cost estimates from the Wolverine Electric Company tripled from the time of the original bid to seven years later when detailed cost information was available. Therefore, it was not recommended by the AEC to move forward with construction [JCAE 1963]. PP&L later informed the AEC that results of a technical feasibility study had indicated insurmountable challenges with the time and resources available; accordingly, the AEC ended negotiations on the PP&L homogeneous reactor in the late 1950s [NSF 1977a, JCAE 1963]. The following reactors are discussed further:

- Homogeneous Reactors (Aqueous):
  - A series of test reactors at LANL
  - Homogeneous Reactor Experiment 1 (HRE-1) at ORNL
  - Homogeneous Reactor Experiment 2 (HRE-2) at ORNL
- Molten Salt Reactors
  - Aircraft Reactor Experiment (ARE) at ORNL
  - Molten Salt Reactor Experiment (MSRE) at ORNL

3.10.1 Low Power (LOPO), High Power (HYPO), Super Power (SUPO), and Los Alamos Power Reactor Experiment (LAPRE) - Test Reactors

LANL designed, constructed, and operated three experimental homogenous fuel reactors, which were referred to using the codename "Water Boilers", due to their supporting role in the Manhattan project. They were referred to as LOPO (low power, operating at nearly zero power), HYPO (high power), and SUPO (super power) [LANL 2004]. LOPO was a homogeneous liquid-fueled reactor with 14% enriched U-235 fuel as uranyl sulfate dissolved in water. LOPO went critical in May 1944, was used for a series of reactivity studies, and was dismantled soon thereafter to make way for the scaled-up version, HYPO, that began operating in December 1944. HYPO used enriched uranyl nitrate dissolved in an aqueous solution; it could be operated at power levels up to 5.5 kWth and thus could provide a stronger source of neutrons needed for cross-section measurements and other studies [Bunker 1983]. By 1950, HYPO was converted to SUPO to enable a range of 35-45 kWth output and continued to operate until 1974 [LANL 2004, ORNL 2009].

LOPO, HYPO, and SUPO were precursors to the Los Alamos Power Reactor Experiment (LAPRE) project. LANL designed and constructed two small reactors, LAPRE-1 and LAPRE-2, between 1955 and 1960 [JCAE 1957a]. LAPRE-1 experiments began in mid-February of 1956 with initial operation set at 20 kWth and then restarted in October of the same year at 160 kWth. After a couple of hours of operation at the second power level, operations were ceased due to detected radioactivity in the steam line caused by corrosion-related failures of the gold-plated stainless steel heat exchangers [DOE 2016, Widner 2009]. LAPRE-1 was shut down while LAPRE-2 was undergoing construction.

LAPRE-2 construction began in May 1956 and operated from February to May 1959 at 800 kWth. Research continued LAPRE-2 using highly-enriched uranium fuel solution dissolved in concentrated phosphoric acid and light water as a coolant and moderator. Both LAPRE-1 and

LAPRE-2's fuel solution temperature and superheated steam output was controlled by setting the uranium concentration and the position of an adjustable control rod. A recurring problem encountered for both reactors was that of achieving satisfactory fuel containment due to corrosion of the primary heat exchanger by high-temperature phosphoric acid. After a contaminated steam was detected in the steam supply, LAPRE-2 was shut down in May 1959 and the LAPRE program came to close in 1960 [Bunker 1983].

3.10.2 Homogeneous Reactor Experiments (HRE) – Test Reactor

By the time that the concept of an aqueous homogeneous reactor concept had been demonstrated at LANL with its three experimental reactors, R&D at ORNL had begun for civilian use of the technology. ORNL initiated a review of the homogenous reactor concept in the late 1940s which led to the design and construction of the first Homogenous Reactor Experiment (HRE-1) completed in 1952 [ORNL 2009]. HRE-1 was designed at 1 MWth but attained a 1.6 MWth heat output during its operational lifetime utilizing the uranyl sulfate aqueous fuel solution and heavy water acting as both moderator and reflector. The electrical power output for HRE-1 was 140 kWe and was considered by the AEC as a "very small, uneconomic, experimental power plant" [JCAE 1954]. Construction of HRE-1 was estimated to cost around $1.1 million in 1957 dollars [JCAE 1957a]. After two years of operation, HRE-1 was shut down in 1954. An ORNL summary of the capabilities and characteristics of the HRE technology is taken from [ORNL 2009]:

- A small homogeneous reactor can be operated reliably at a significant temperature and pressure;
- Leak tightness can be maintained;
- Nuclear stability is excellent; and
- Maintenance is feasible in spite of the circulating fission products.

HRE-1's operation led to further R&D inquiries associated with a larger scale reactor, the HRE-2, which was designed to be rated at 5 MWth. The spherical core of HRE-2 was divided into two zones, a blanket circulating heavy water as a reflector and moderator and the main core circulating the aqueous fuel fluid. Instead of uranyl sulfate in light water, heavy water was used as the solvent in the fuel mixture. Power output was maintained at the same 140 kWe, and a chemical processing plant was added to remove fission products from the fuel stream [JCAE 1954, ORNL 2009]. Installation of the HRE-2 began in July 1952 and first criticality was reached in early 1958. The estimated HRE-2 cost in 1957 was over $3 million [JCAE 1957a]. HRE-2 suffered operational upsets due to corrosion leading to reduction of flow in the reactor vessel and unanticipated high-frequency power cycles caused by uranium coming out of solution. Other phenomena that were investigated were the boiling of the aqueous fuel solution within the main core and low core fluid velocity resulting from entrapment of solid particles in the core and low heat-transfer coefficients [ORNL 2009]. After HRE-2 was shut down in 1961, plans for HRE-3, a 65 MWth thorium-based breeder reactor, did not materialize [ORNL 1957].

3.10.3 Aircraft Reactor Experiment (ARE) –Test Reactor

As a result of military interest in this unique application, the first MSR demonstration was the Aircraft Reactor Experiment (ARE), a short duration proof-of-concept test [Bettis 1957, Serp 2014]. The ARE was developed as part of the Aircraft Nuclear Propulsion (ANP) program initiated in 1946 by the US Air Force targeting propulsion systems capable of powering aircraft for extended strategic operations without refueling. The US Air Force contracted Fairchild Engine and Airplane Corporation to perform a technology feasibility study in 1946 while in 1948 the AEC separately requested that Massachusetts Institute of Technology (MIT) perform an independent evaluation of the project's cost and schedule. Conclusions from both studies were cautiously optimistic, and MIT reported that costs could reach $1 billion over a 15-year duration [ORNL 2009, Stoffel 2000].

On-ground testing of the ARE at ORNL occurred in 1954 and lasted for 100 hours. The reactor operated up to 2.5 MWth with steady state outlet temperatures around 860°C. The ARE used Inconel as the primary structural material, beryllium-oxide as the moderator, sodium as the primary coolant, helium as the secondary coolant, and NaF-ZrF4-UF4 as the fuel salt [ORNL 2009]. The one test run demonstrated stable, self-regulating MSR operation at very high temperatures. Reliable operations of the ARE were challenging during loading of the fuel concentrate, which required clearing the transfer lines. The cause of this plugging was the melting point difference of the fuel concentrate and the carrier salt that led to freezing of the fuel concentrate. The ANP program ended in 1961 when the DoD determined that the mission was not required. [Stoffel 2000].

3.10.4 Molten Salt Reactor Experiment (MSRE) - Test Reactor

The Molten-Salt Reactor Experiment (MSRE) at ORNL followed the ARE and was designed to operate at larger scale for a longer period of time to evaluate the technology for use in power reactors [Rosenthal 1968, ORNL 2001]. A primary objective of the MSRE was to evaluate the chemical compatibility of the three materials that made up the reactor: fuel salt, unclad graphite moderator, and a high-nickel alloy. A secondary objective was to demonstrate operation with uranium-233 fuel. The MSRE started in mid-1965 with enriched uranium fuel and was rated at a 7.5 MWth heat output. Examination of reactor components and materials during the test indicated compatibility of the salt-moderator-metal alloy system for the limited duration of the test. After the enriched uranium was removed from the fuel salt and replaced by uranium-233 (recovered from commercial nuclear fuel), the MSRE was restarted in 1968 and became the first reactor to operate on uranium-233. The MSRE operated for 20,000 hours. The next planned phase in MSR development was the construction of a larger molten-salt reactor demonstrating thorium breeding to uranium-233 and associated chemical processing required [Rosenthal 1968]. However, budget constraints led the AEC to prioritize and select a single breeder reactor concept for further development in the US (the CRBR). Consequently, all major MSR developmental efforts ended in 1972, although minimal R&D work continued through the 1970s.

Table 3-9
Summary of US Aqueous Homogeneous Reactor and Molted Salt Reactor Development.

Plant	Location	Owner	Initial Criticality Date [Operating dates]	Size [MWe, MWth]	Type of Facility [Experiment/Test, Small-Scale Demonstration, Large-Scale Demonstration, Commercial]	Stage of development
LOPO	LANL site (NM)	Gov't (AEC)	[May 1944 – Dec 1944]	~0 kWth	Experiment/Test	C (constructed, limited ops)
HYPO	LANL site (NM)	Gov't (AEC)	[1944-1950]	5.5 kWth	Experiment/Test	C (constructed, limited ops)
SUPO	LANL site (NM)	Gov't (AEC)	[1950-1974]	35-45 kWth	Experiment/Test	C (constructed, limited ops)
LAPRE-1	LANL site (NM)	Gov't (AEC)	[1956-1959]	2-160 kWth	Experiment/Test	C (constructed, limited ops)
LAPRE-2	LANL site (NM)	Gov't (AEC)	[Feb 1959-May 1959]	800 kWth	Experiment/Test	C (constructed, limited ops)
HRE-1	ORNL site (TN)	Gov't (AEC)	[1952-1954]	1.6MWth 140 kWe	Experiment/Test	C (constructed, limited ops)
HRE-2	ORNL site (TN)	Gov't (AEC)	[1958-1961]	5 MWth 140 kWe	Experiment/Test	C (constructed, limited ops)
ARE	ORNL site (TN)	Gov't (AEC)	[1954]	2.5 MWth	Experiment/Test	C (constructed, limited ops)
MSRE	ORNL site (TN)	Gov't (AEC)	[1965-1972]	7.5 MWth	Small Demo	C (constructed, limited ops)

3.11 Heavy Water Reactors (HWR) in US

Although heavy water reactors were used in the US plutonium production program at Savannah River, the U.S. lagged behind Canada's RD&D activities to develop the HWR for civilian power applications. As Canada finalized plans for NPD and Douglas Point in the latter half of the decade, the U.S. began the solicitation process to begin exploring the HWR concept for electric power production in the late 1950s. The HWR concept pursued was the Carolina-Virginia Tube Reactor (CVTR) which is discussed below.

3.11.1 Carolina-Virginia Tube Reactor (CVTR)

In the second round of the PDRP, the AEC was particularly interested in developing the heavy water reactor (HWR) that would utilize natural uranium fuel and the homogenous reactor (as described in Section 5.3). The Carolinas Virginia Nuclear Power Associates (CVNPA), led by the Carolina Power & Light Company (CP&L)[1], submitted its proposal to design, construct, and

[1] Three other utilities were part of the CVNPA: Duke Power Company, South Carolina Electric and Gas Company, and Virginia Electric and Power Company.

operate a HWR plant at Parr, South Carolina, called the Carolina-Virginia Tube Reactor (CVTR) in August 1957 [IAEA 2002]. Initial planning and design of the CVTR dates back to approximately 1955 when the AEC granted access for certain CP&L employees to receive classified information on nuclear energy, including the reactor concepts then in use at the AEC Hanford and Savannah River sites. Construction commenced in 1960 and operations occurred from 1963 to 1967 [IAEA 2002, Morris 1993].

As part of the agreement with the AEC, CVNPA established that Westinghouse would be the contractor for a 19 MWe pressurized heavy water-moderated, tube-type reactor [IAEA 2002, Morris 1993]. Stone & Webster designed the remainder of the electrical portion of the power plant [ACRS 1978]. Under the agreement, CVNPA would put up $22 million for capital costs while the AEC would pay $15 million for R&D and fuel use charges [Morris 1993]. The CVTR was a pressurized heavy-water cooled and moderated reactor system run with uranium in zircaloy tubes. Additionally, the CVTR incorporated an oil fired superheater to upgrade the quality of the steam being fed to the turbine. The reactor operated for four years without major issues problems and was shut down after that time when sufficient operating information on the design had been obtained [IAEA 2002].

Table 3-10
Summary of US Aqueous Homogeneous Reactor and Molted Salt Reactor Development.

Plant	Location	Owner	Initial Criticality Date [Operating dates]	Size [MWe, MWth]	Type of Facility [Experiment/Test, Small-Scale Demonstration, Large-Scale Demonstration, Commercial]
CVTR	Parr (SC)	Gov't (AEC)	[1963-1967]	19MWe	Small Demo

3.12 Summary and Observations

Reactor technologies that were fully commercialized are summarized below, followed by reactor types that did not reach the commercialization stage in the U.S. (e.g., sodium-cooled reactors). Preliminary observations of the general trends that could be gleaned from the literature are presented for both categories of reactor technologies.

3.12.1 Commercialized Reactor Technologies (PWRs, BWRs, CANDU, MAGNOX)

Precursor Test Reactors

In the commercialized technologies studied, test reactors were built and operated in the 1940s and 1950s to study the harnessing of nuclear fission for non-weapons purposes. The primary focus was to demonstrate controlled, sustainable nuclear heat generation as well as to observe material behavior in high neutron flux environments. Such test reactors either generated no electricity or very small amounts, up to 5 MWe. In most cases, these early reactors were fully sponsored and directed by governments. In the U.S., the 1954 modifications to the Atomic Energy Act (in part) enabled a privately initiated test reactor to be built and started up in 1957 (VBWR), which illustrates the early interest that both government and industry had in development of commercial nuclear power technology – the government was willing to lift restrictions on private use of nuclear materials, and industry was willing to invest private funds to further technology development.

Small Demonstration Reactors

The proofs-of-principle physically demonstrated by the precursor test reactors led to government decisions to advance to the next step of technological development - small demonstration reactors. This paper defines this decision point as the starting place for the commercialization lead time. For the PWR technology, this corresponded to the U.S. government decision, in 1954, to award a contract to build the 60 MWe Shippingport Atomic Power Station. For the BWR technology, it corresponded to the U.S. government decision to enter into contract negotiations, in 1956, for the 22 MWe Elk River Reactor. For the HWR (CANDU) technology, this corresponded to the 1955 contractor selection for the 20 MWe NPD reactor. Lastly, the start of the commercial lead time for the MAGNOX GCR corresponds to the 1953 establishment of the UK's civil nuclear power program. At each of these points in time, the focus of technology development had shifted from precursor test reactors to small demonstration reactors which were needed to prove that nuclear heat could be generated safely and consistently and be used to reliably produce electrical power. Each of these small demonstration reactors was less than 100 MWe.

In Canada and the UK, the small demonstration reactor phase of technology development was funded and guided by the government, but involved private contractor participation. The same was true for some of the projects in the U.S; however, the U.S. government also opened the door for expanded government and industry collaboration via the AEC's Power Demonstration Reactor Program (PDRP) whereby private industry was able to receive technical, financial, and nuclear materials support from the federal government for reactor proposals developed by industry. It was also in this timeframe that the U.S. enacted legislation to lessen the burden on private utilities of nuclear power plant accident liability (Price Anderson Act of 1957). Therefore, much was happening in the U.S. to engage private enterprise in the development and deployment of commercial nuclear power. The spectrum of U.S. government and industry involvement in small PWR and BWR demonstration reactors consisted of: (1) government controlled and operated (Shippingport, EBWR), (2) industry owned and operated (Saxton Nuclear Experimental Reactor and the VBWR, to the degree it generated intermittent power), and (3) government and industry collaboration (5 PDRP reactor projects). The last category included two BWR integral nuclear superheater concepts, built in parallel and built (in retrospect) without the benefit of adequate proof-of-principle precursor testing. The implementation problems encountered at BONUS and Pathfinder effectively ended any industry interest in pursuing the integral superheater concept–reinforcing the prudency of the step-wise, sequential technology development process.

Large Demonstration Reactors

Subsequently, but also sometimes overlapping in time, the early 1960's saw the startup of larger sized demonstration reactors – the next step in the progression to large commercial-scale power reactors. In the UK and Canada, which had tighter control on the development processes, the transition from small to large scale was strictly sequential. The U.S. was approximately so, but policy changes supporting privatization (and less central planning by government) yielded exceptions. For instance, the privately initiated and funded Dresden Unit 1 BWR power reactor (200 MWe) started 2 to 3 years earlier than the 5 BWR PDRP small demonstration reactor projects. The fully private Dresden venture was built upon test reactor experience at BORAX I-III and the experience of operating the private VBWR test reactor, which had a very low power output of 5 MWe. As Dresden operated between 1960 and 1978, the leap in power capacity from VBWR (5 MWe) to Dresden (200 MWe) could be looked upon as a partial refutation of the need for step-wise, sequential technology development; however, it can also be noted that Dresden was able to draw on substantial testing and experience with LWR technology (STR, SAPS), overall, and parallel work ongoing at the EBWR at ANL.

Ongoing Testing and Experiments

It is important to point out that additional test reactor work occurred throughout the development and construction of the small and large demonstration reactors in the U.S., U.K., and Canada; further, such research continued into the years of large-scale commercial nuclear power plant operations and beyond. We can refer to the initial proof-of-principle test reactors that preceded the demonstration reactors as precursors and acknowledge the need for an ongoing parallel track of experiments run for such purposes as: development of improved computational modeling, investigation of emerging reactor phenomena, testing for desired plant modifications, and investigation of proposed operational and safety improvements. Test facilities such as TREAT and LOFT can be viewed in this regard, as can continued construction of private sector test reactors and nuclear testing facilities in the U.S. In this way, the U.S. government,with industry support and involvement in nuclear power related testing, has continued to tackle RD&D problems of importance.

Commercial Reactors

In this study, the technology development period ended when the first privately funded/owned commercial-scale reactor started operations. The commercial-scale reactor is sized to be economically competitive and typically has on the order of twice the power generating capacity of the large scale demonstration reactors. For the U.S. PWR, this was signified by the startup of the RE Ginna facility (580 MWe) in 1969. The time that had elapsed since the decision to pursue a small PWR demonstration reactor was roughly 15 years; thus, the PWR lead time can be considered to be on the order of 15 years. In the same fashion, the U.S. BWR lead time can be considered to be roughly 13 years, as the initial commercial BWR, Oyster Creek, started operations in 1969. In Canada, the first Pickering A unit began operations in 1973, making the HWR CANDU lead time approximately 16 years. And, as Hinkley Point 1 began operations in 1965, the lead time for the UK CGR is show to be approximately 12 years. These lead times are graphically depicted in Figure 3-2.

It is useful to compare these historical lead times to often quoted lead time estimates ranging from 30 to 50 years for the commercialization of new sources of large-scale, low-carbon, power generation options, including advanced nuclear reactor technologies. Nuclear industry history, as recounted in this paper, records lead times less than half these current day estimates. History suggests that at least from a technical perspective, the timeframe for deployment of new nuclear technologies is overly pessimistic. It should be noted that this time frame was based on a firm government decision to pursue a small demonstration reactor for the technology and significant pre-cursor R&D having been completed in two or more test reactors.

Also, this conclusion needs to be predicated on the understanding that certain conditions that existed in the past may need to be readapted in a form useful for the current day. Nuclear power development in the U.S. benefited from three characteristics of government involvement. First, federal government policy supported peaceful applications of nuclear technologies, particularly the development and deployment of commercial nuclear power. Commensurate with that policy, the U.S. government shared relevant technical data and utilized the nation's nuclear science expertise and facilities to conduct necessary precursor testing of reactor concepts. Further, government provided financial support to industry in the form of government-sponsored R&D, provision of nuclear materials, cost-sharing for facility construction, and utility indemnification from postulated nuclear accident liability. Commercialization in Canada and the U.K. benefitted from similar levels of government support.

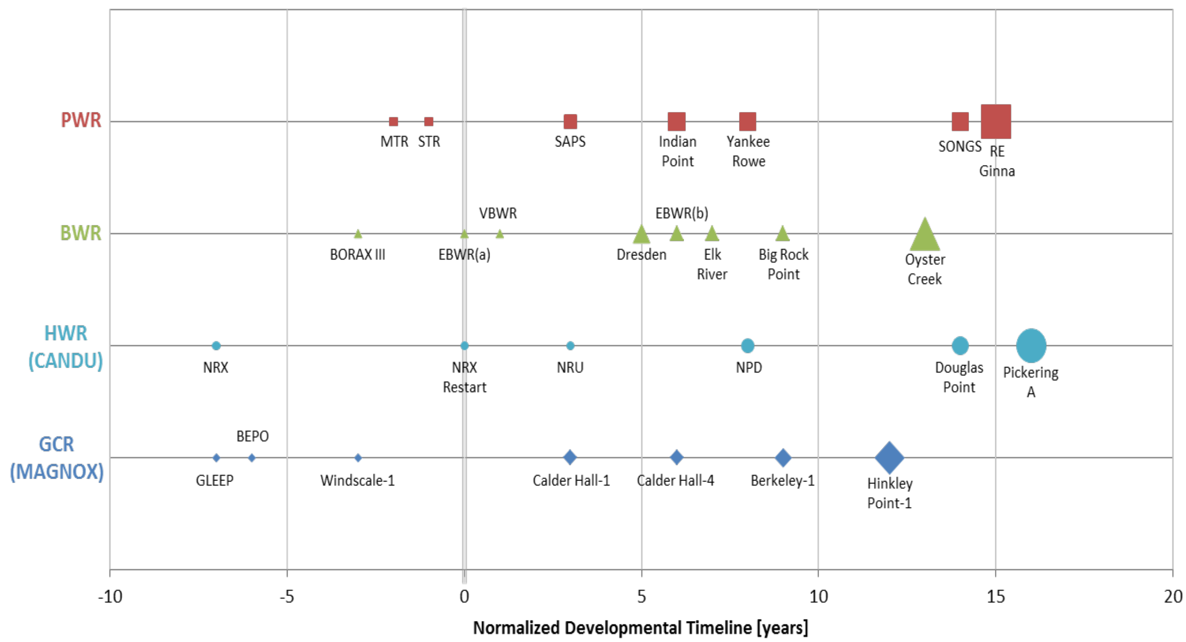

Figure 3-2
Normalized Reactor Developmental Timelines.

<u>Figure 3-2 Notes:</u>

t=0 corresponds to decision for Demonstration Reactor	Lead Times:
PWR – 1954 contract for SAPS	U.S. PWR ~15 yrs
BWR – 1956 negotiations for Elk River	U.S. BWR ~13 yrs
CANDU – 1955 contractor selection for NPD	Canada CANDU ~16 yrs
GCR – 1953 announcement for civil nuclear power program	UK GCR ~12 yrs

In general, government and private industry partnership supported the gradual increase of the capacity of each reactor technology over time. Thus, it was not simply one facility, but a process of gradual capacity enlargement—through multiple facilities—that fostered the technological maturation of each reactor type (and its associated fuel cycle) into a commercially viable state. This progression of design sizes is shown in Figure 3-3 through Figure 3-6 for each technology. The same information is provided in tabular form in Table 3-11.

Areas of participation by the government and private industry are shown in Table 3-12 for the four scales of reactors as they progressed to the commercial state. Scanning the columns from the test reactor phase to initial commercial reactors (left to right), the gradation from mostly government support (designated as "G-") to increasing levels of private industry support ("PS-") is evident.

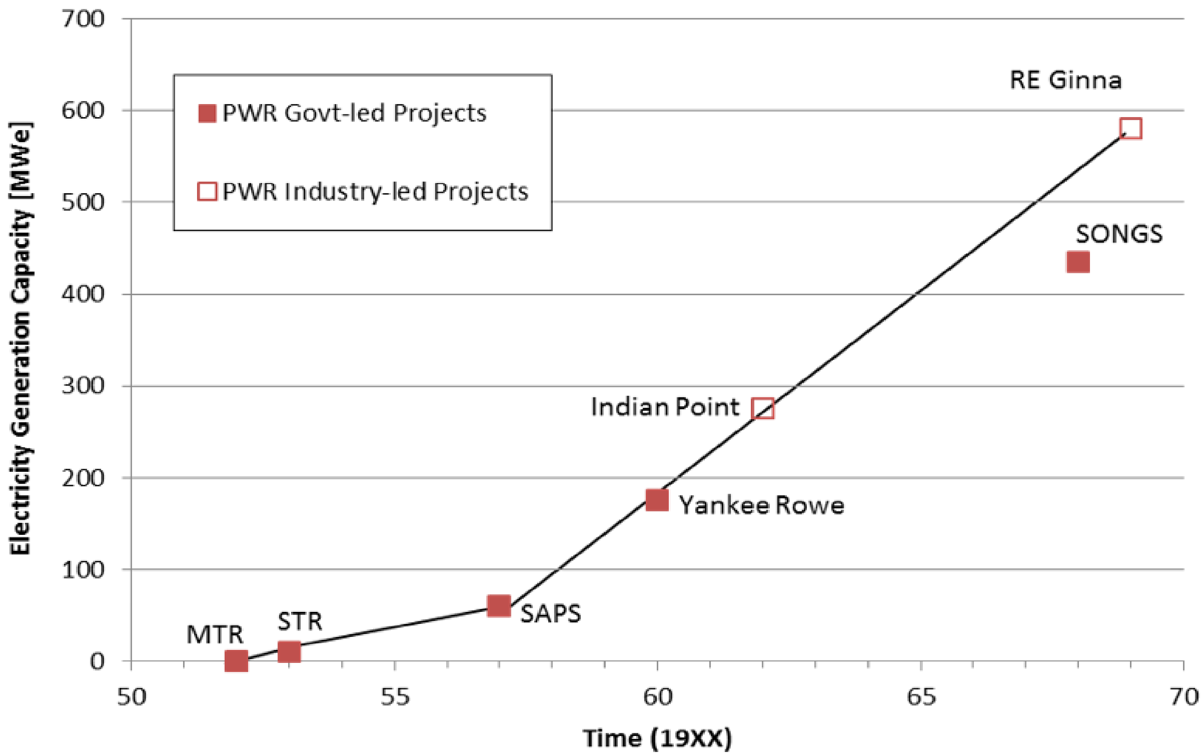

Figure 3-3
PWR Development Timeline.

<u>Figure 3-3 Notes:</u>

Data points designate operational start dates.

MTR = Materials Testing Reactor [1952, 0 MWe]

STR = Submarine Thermal Reactor [1953, 10 MWe]

SAPS = Shippingport Atomic Power Station [1957, 60 MWe]

Yankee Rowe [1960, 175 MWe]

Indian Point* was a commercial reactor [1962, 275 MWe]

SONGS = San Onofre Nuclear Generating Station [1968, 435 MWe]

RE GINNA [1969, 580 MWe]

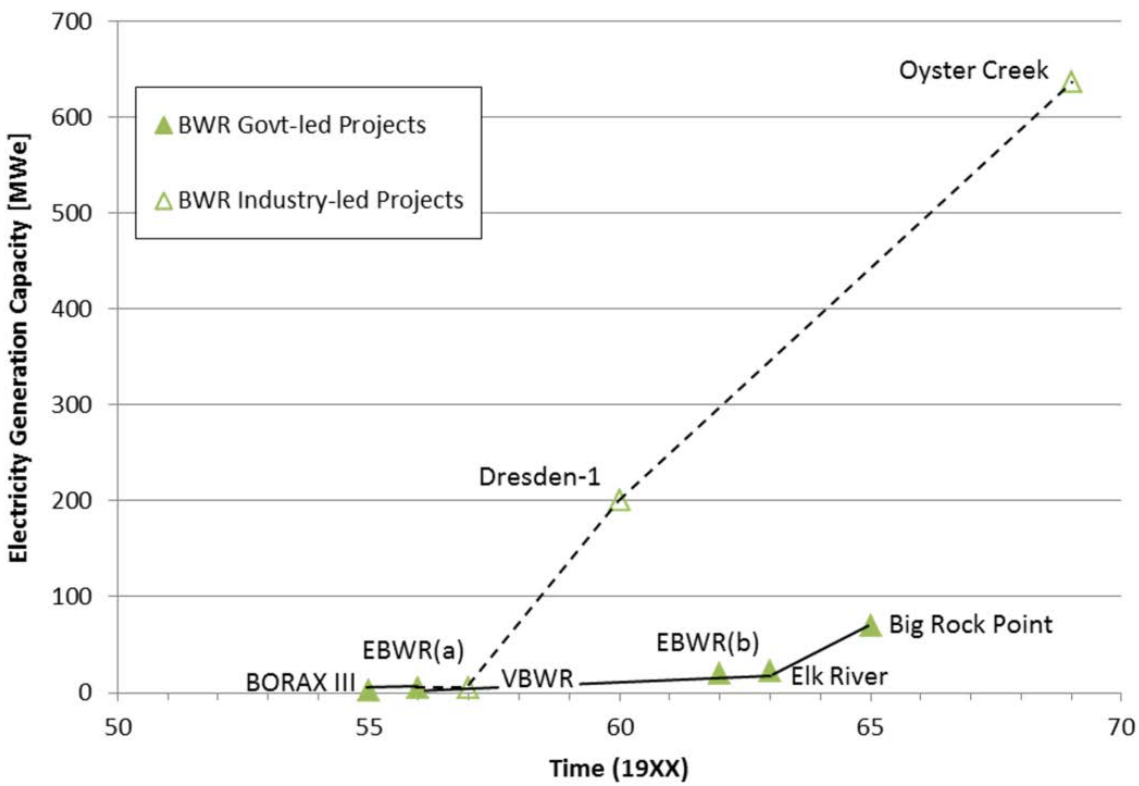

Figure 3-4
BWR Development Timeline.

Figure 3- 4 Notes:

Data points designate operational start dates.

BORAX III = Boiling Water Reactor Experiment III [1955, 2 MWe]

EBWR(a) = Experimental Boiling Water Reactor at the time of initial operations [1956, 5 MWe]

EBWR(b) = Experimental Boiling Water Reactor after increasing electrical capacity [1962, 20 MWe]

Elk River [1963, 22 MWe]

Big Rock Point [1965, 70 MWe]

VBWR = Vallecitos Boiling Water Reactor [1957, 5 MWe]

Dresden-1 [1960, 200 MWe]

Oyster Creek [1969, 636 MWe]

This figure does not show the discontinued technology development on the integral, nuclear superheater BWR concept.

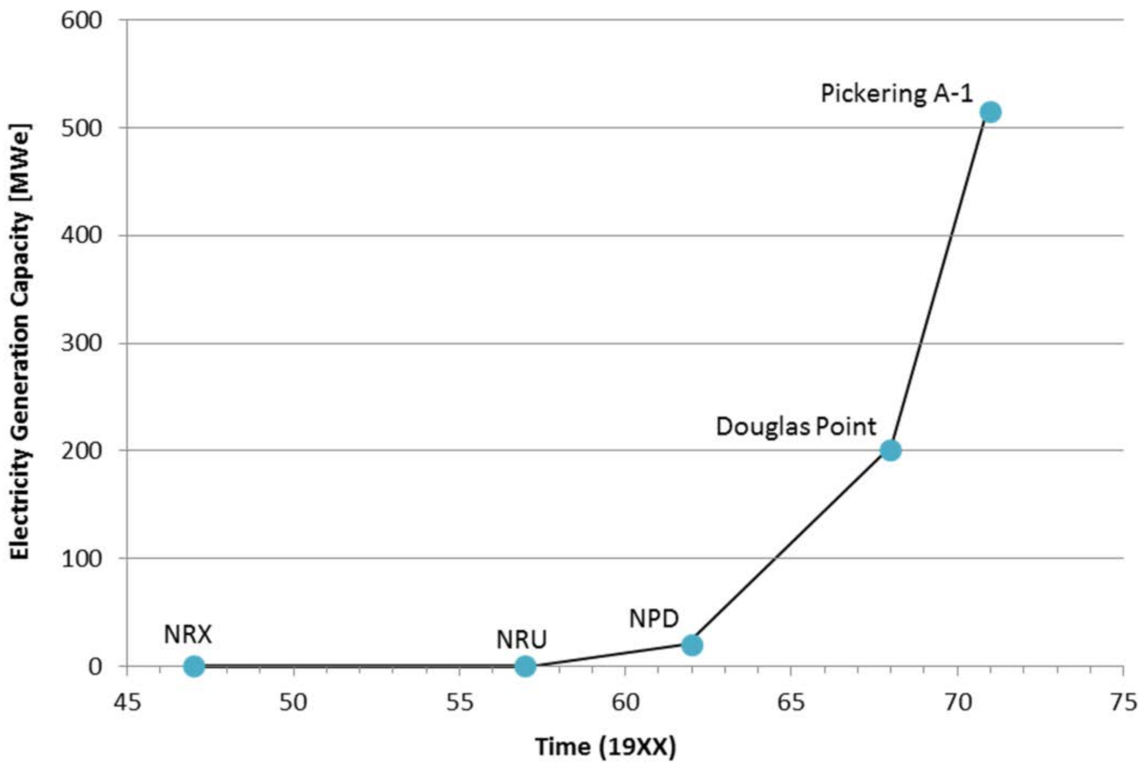

Figure 3-5
HWR (CANDU) Development Timeline.

<u>**Figure 3-5 Notes**</u>:

Data points designate operational start dates.

NRX = National Research Experimental Reactor [1947, 0 MWe]

NRU = National Research Universal Reactor [1957, 0 MWe]

NPD = Nuclear Power Demonstration [1962, 20 MWe]

Douglas Point [1968, 200 MWe]

Pickering A-1 [1971, 515 MWe]

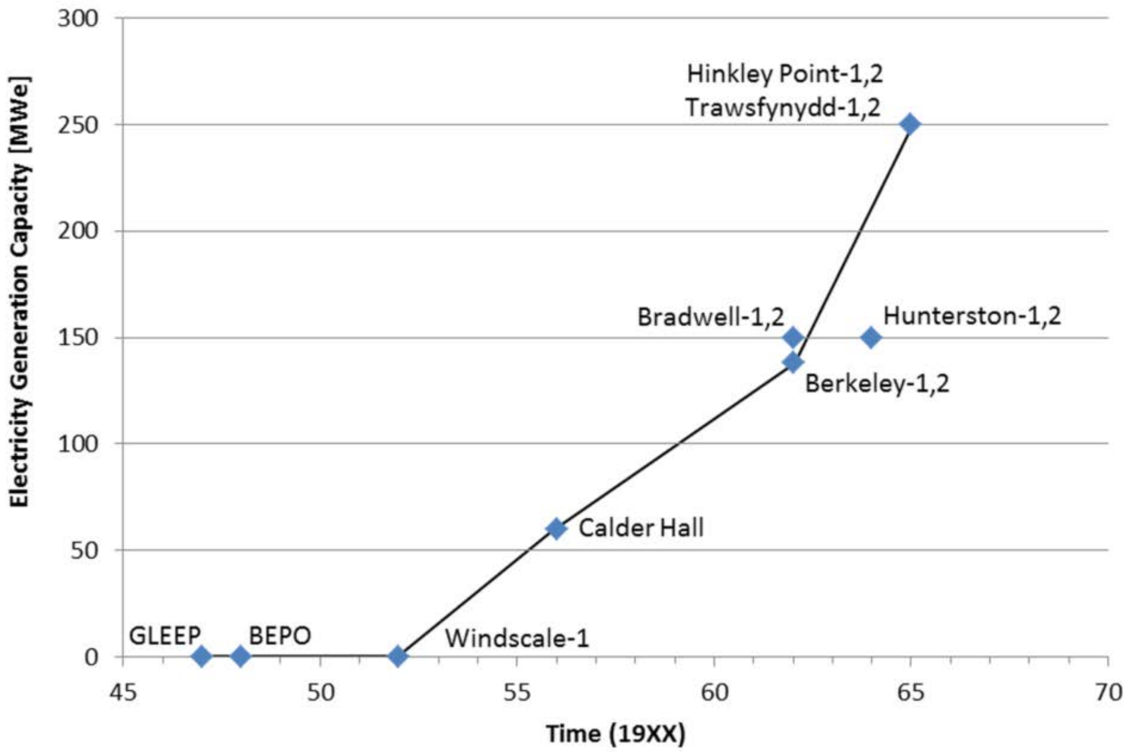

Figure 3-6
GCR (MAGNOX) Development Timeline.

Figure 3-6 Notes:

Data points designate operational start dates.

GLEEP = Graphite Low Energy Experimental Pile [1947, 0 MWe]

BEPO = British Experimental Pile '0' [1948, 0 MWe]

Windscale-1,2 [1952, 0 MWe]

Calder Hall-1 [1956, 60 MWe]

Berkeley-1,2 [1962, 138 MWe]

Bradwell-1,2 [1962, 150 MWe]

Hunterston-1,2 [1964, 150 MWe]

Hinkley Point-1,2 [1965, 250 MWe].

Table 3-11
Reactor Development Comparison Table.

Technology (% Installed)	Precursor Test Reactors	Small Demonstration Reactors	Large Demonstration Reactors	Initial Commercial Reactor(s) & Presently Installed Base
PWR (69%)	MTR [1952, 0 MWe] STR [1953, 0 MWe]	SAPS [1957, 60 MWe]	Yankee Rowe [1960, 175 MWe] San Onofre [1968, 435 MWe] Indian Point [1962, 275 MWe]	RE Ginna [1969, 580 MWe] More than 260 operating reactors in ~27 countries
BWR Government-led Projects	BORAX I-III [1953-56, 2 MWe] EBWR(a) [1956, 5 MWe]	EBWR(b) [1962, 20 MWe] Elk River [1963, 22 MWe] Big Rock Point [1965, 70 MWe]		
BWR Industry-led Projects (15%)	VBWR [1957, 5 MWe]		Dresden-1 [1960, 200 MWe]	Oyster Creek [1969, 636 MWe] More than 55 operating reactors in 10 countries
CANDU (11%)	ZEEP [1945, 0 MWe] NRX [1947, 0 MWe] NRX Restart [1954, 0 MWe] NRU [1957, 0 MWe]	NPD [1962, 20 MWe]	Douglas Point [1968, 200 MWe]	Pickering A-1 [1971, 515 MWe] 12 CANDUs and 13 CANDU-derivatives operating in 6 countries
MAGNOX (4%)	GLEEP [1947, 0 MWe] BEPO [1948, 0 MWe] Windscale-1,2 [1951-52, 0 MWe]	Calder Hall-1,2,3,4 [1956-59, 60 MWe] Chapelcross-1,2,3,4 [1956-60, 60 MWe]	Berkeley-1,2 [1962, 138 MWe] Bradwell-1,2 [1962, 150 MWe] Hunterston-1,2 [1964, 150 MWe]	Hinkley Point-1,2 [1965, 250 MWe] 8 additional 1956-58 MAGNOX (250-290 MWe) Wylfa-1,2 [1971, 590 MWe] 14 Advanced GCRs operating (follow-up of the MAGNOX)

Table 3-12
Participation Areas of Government and the Private Sector for Reactor Technologies.

Participation Area	Test Reactors	Small Demonstration Reactors	Large Demonstration Reactors	Initial Commercial Reactor
Site Acquisition	G-MTR, G-Mk-I, G-BORAX, G-EBWR, G-UK, PS-NRX	PS-SAPS, PS-ER, PS-BRP, PS-LACBWR, PS-NPD, G-CH**	PS-YR, PS-HN, G-SONGS, PS-DP, PS-B1	PS-REG PS-P
Reactor Owner	G-MTR, G-Mk-I, G-BORAX, G-EBWR, G-NRX, G-UK	G-SAPS, G-ER, G-BRP*, G-LACBWR*, G-NPD, G-CH**	G-YR*, PS-HN, G-SONGS*, G-DP, PS-B1	PS-REG PS-PA
Power Plant Owner	G-MTR, G-Mk-I, G-BORAX, G-EBWR	PS-SAPS, PS-ER, PS-BRP, PS-LACBWR, PS-NPD, G-CH**	PS-YR, PS-HN, G/PS-DP, PS-B1	PS-REG PS-PA
Pre-Construction R&D	G-MTR, G-Mk-I, G-BORAX, G-EBWR, G-NRX, G-UK	G/PS-SAPS, G/PS-ER, PS-BRP***, G/PS-NPD, G-CH	G/PS-YR, G/PS-DP, G/PS-B1	PS-REG G/PS-PA
Post-Construction R&D	G-MTR, G-Mk-I, G-BORAX, G-EBWR, G-NRX, G-UK	G/PS-SAPS, G/PS-ER, PS-BRP***, PS-LACBWR, G/PS-NPD, G-CH	G/PS-YR, G/PS-DP, PS-B1	PS-REG G/PS-PA
Reactor Plant Design	G-MTR, G-Mk-I, G-BORAX, PS-EBWR, G-NRX, G-UK	G/PS-SAPS, G/PS-ER, PS-BRP, PS-LACBWR G/PS-NPD, G/PS-CH	PS-YR, G/PS-HN, G/PS-DP, G/PS-B1	PS-REG G/PS-PA
Power Plant Design	G/PS-Mk-I, G-BORAX, PS-EBWR, PS-NRX,	PS-SAPS, PS-ER, PS-BRP, PS-LACBWR PS-NPD, G/PS-CH	PS-YR, PS-HN, PS-DP, G/PS-B1	PS-REG G/PS-PA
Fuel Design	G-MTR, G-Mk-I, G-BORAX, G-EBWR, G-NRX, G-UK	G/PS-SAPS, G/PS-ER, PS-BRP, PS-LACBWR, G/PS-NPD, G/PS-CH	PS-YR, PS-HN, G-DP, G/PS-B1	PS-REG G/PS-PA
Fuel Fabrication and/or Supply	G-MTR, G-Mk-I, G-BORAX, G-EBWR, G-NRX, G-UK	G-SAPS, G-ER, G-BRP, G- LACBWR, G-NPD, G-CH	G-YR, G/PS-HN, G-SONGS, G-DP, G-B1	PS-REG G/PS-PA
Rate Assistance (steam and/or electricity)	PS-NRX	PS-SAPS, G-LACBWR, G-CH	G-DP, G-B1	G/PS-PA**
Reactor Operator	PS-MTR, PS-Mk-I, G-EBWR, PS-NRX, G-UK	PS-SAPS, PS-ER, PS-BRP, PS-LACBWR, PS-NPD, G-CH,	PS-YR, PS-DP, PS-B1	PS-REG PS-PA
Power Plant Operator	G/PS-Mk-I, G-BORAX, G-EBWR, PS-NRX	PS-SAPS, PS-ER, PS-BRP, PS-LACBWR, PS-NPD, G-CH	PS-YR, PS-HN, PS-DP, PS-B1	PS-REG PS-PA
Reactor Plant Constructor	G/PS-MTR, G/PS-Mk-I, G-BORAX, PS-EWBR, PS-NRX, G-UK,	G/PS-SAPS, G/PS-ER, G-LACBWR, PS-NPD, G/PS-CH	PS-YR, PS-HN PS-DP, G/PS-B1,	PS-REG PS-PA
Power Plant Constructor	G/PS-MTR, G/PS-Mk-I, G-BORAX, PS-EWBR	PS-SAPS, PS-ER, PS-LACBWR, PS-NPD, PS-CH	PS-YR, PS-HN, PS-DP, PS-B1	PS-REG PS-PA

Table 3-12 Notes:

Test Reactors:

G = government (used throughout the table);

PS = private sector (used throughout the table);

Mk-I = Mark I;

MTR = Materials Testing Reactor;

BORAX = Boiling Water Reactor Experiment;

EBWR = Experimental Boiling Water Reactor;

NRX = Nuclear Reactor Experiment (Canada);

UK = United Kingdom test reactors;

Small Demonstration Reactors:

SAPS = Shippingport Atomic Power Station;

ER = Elk River BWR;

BRP = Big Rock Power BWR;

NPD = Nuclear Power Demonstration (Canada);

CH= Calder Hall (United Kingdom)

*LACBWR (La Crosse BWR) = AEC was the original owner and later sold the plant to DPC and provided them with an operating license.

**CH (Calder Hall) = dual use reactor, also produced plutonium for UK weapons program

***BRP (Big Rock Point BWR) = ownership was later transitioned to the private sector

Large Demonstration Reactors:

YR = Yankee Rowe PWR;

HN = Haddam Neck PWR;

DP = Douglas Point (Canada);

B1 = Berkeley 1 (United Kingdom);

SONGS = San Onofre Nuclear Generating Station;

*YR (Yankee Rowe PWR) = ownership was later transitioned to the private sector

*SONGS (San Onofre Nuclear Generating Station) = ownership later transitioned to private sector

Initial Commercial Reactors:

PA = Pickering A (Canada)

REG = RE Ginna PWR

**PA: Pickering A [Canada] = Because the operating cost of Pickering A was expected to be significantly less than that of a coal station, Ontario Hydro agreed to pay back the government investments from the resulting savings. Over time, the government investments were fully repaid [Whitlock 2016a].

3.12.2 Reactor Technologies Yet to be Commercialized

While the UK and Canadian commercialization programs each focused on a particular reactor technology from the beginning of the development cycle, the U.S. took a more expansive path that allowed and encouraged the parallel consideration of multiple reactor technologies. The expansive approach is reflected in both the history of government-sponsored research and the history of private industry investment in reactor projects. As illustrated by this paper, the initial openness to a broad spectrum of technologies did not seem to be disadvantageous to the U.S. on the world stage in terms of lead time to nuclear power commercialization nor in terms of the timing for the ultimate achievement of this goal. Nonetheless, given the high level of technological complexity inherent in all reactor concepts, and the high-stakes-high-reward nature inherent in nuclear power technology – with attendant need for large financial investment and the particular imperative for a focus on safety - it is perhaps not surprising to see that both the government and the market-place favored a narrowing on the technological path for commercialization. Focus on the implementation of a particular technology can offer advantages in regulatory and oversight processes, consumer and investor confidence, and cost efficiency – a consideration that may be important in the path forward for advanced reactor concepts. For multiple technical and non-technical reasons, the full range of which is beyond the present scope of this paper, the U.S. path to commercialization was narrowed to the PWR and BWR light-water reactor concepts. Therefore, it has not been the purpose of this paper to pinpoint why the several non-LWR technologies, developed as part of the PDRP, did not also achieve commercialization but merely to document key points of U.S. non-LWR development history.

3.12.3 Important Caveats

When drawing on data from the 1950's and 1960's, it must be noted that the nuclear technology development landscape was different then. As this white paper is developed further, we will describe the impact that such differences had on the pace of technology development; however, here we will at least list some of the salient differences.

- During this time period the U.S. and the (then) Soviet Union were engaged in what has been described as "the Cold War", a period of international tension and, importantly completion; this competition extended over a broad scope of public affairs—including the development of civilian nuclear power.
- The development of nuclear power technology was supported by unitary government agencies that combined the missions of technology development (and advocacy) with the environmental and safety regulatory functions.
- In the U.S., changes to the Atomic Energy Act in the years after the PDRP would limit the government role in commercialization of nuclear technology to a degree.

3.13 References

[ACRS 1978] Okrent, D., On the History of the Evolution of Light Water Reactor Safety in the United States. 1978, US Advisory Committee on Reactor Safeguards (ACRS). p. 1116 pages.

[AEC 1957] (AEC), U.A.E.C., Theoretical possibilities and consequences of major accidents in large nuclear power plants: a study of possible consequences if certain assumed accidents, theoretically possible but highly improbable, were to occur in large nuclear power plants. Vol. WASH-740. 1957.

[AEC 1961] US Atomic Energy Commission (AEC), Liquid Metal Fast Breeder Reactors 1948-1961. p. 129 pages.

[AEC 1964] Mueller, H.I., Nuclear Reactors Built, Being Built, or Planned in the United States as of June 30, 1964. 1964, U.S. Atomic Energy Commission. p. 33 pages.

[AEC 1069] McCardell, R.K., D.I. Herborn, and J.E. Houghtaling, Reactivity Accident Test Results and Analyses for the SPERT III-E Core – A Small Oxide-Fueled, Pressurized Water Reactor. 1969, U. S. Atomic Energy Commission (AEC). p. 199 pages.

[AEC 1972] Atomic Energy Commission (AEC), Hewlett, R.G. and F. Duncan, Atomic Shield, 1947-1952. Vol. 2 1972, (URL: http://energy.gov/sites/prod/files/2013/04/f0/Hewlett%20and%20Duncan%20-%20Atomic%20Shield%20%28complete%29.pdf). 747 pages.

[AEC 1989] Hewlett, R.G. and J.M. Holl, Atoms for Peace and War, 1953-1961. 1989, US Atomic Energy Commission (AEC). p. 733 pages.

[AECL 1997] Atomic Energy of Canada Limited (AECL), Canada Enters the Nuclear Age: A Technical History of Atomic Energy of Canada Limited Vol. (URL: http://pbadupws.nrc.gov/docs/ML0303/ML030360478.pdf). 1997: McGill-Queen's University Press.

[ANL 1957] Argonne National Laboratory (ANL), Experimental Boiling Water Reactor (EBWR). 1957. p. 250 pages.

[ANL 1958] Brittan, R.O., Analysis of the EBR-I Core Meltdown. 1958, Argonne National Laboratory (ANL). p. 14 pages.

[ANL 1991] Sackett, J.I., Operating and Test Experience of EBR-II. 1991, Argonne National Laboratory (ANL). p. 19 pages.

[ANL 2006] Argonne National Laboratory (ANL). Experimental Boiling Water Reactor (EBWR). 2006 Last Revised October 30, 2006 [cited Accessed Jne 2, 2016; Available from: http://www.dd.anl.gov/projects/ebwr.html.

[ANL 2016a] Argonne National Laboratory (ANL). Reactors Designed by Argonne National Laboratory. 2016 Last Revised June 1, 2016 [cited Accessed June 2, 2016; Available from: http://www.ne.anl.gov/About/reactors/lwr3.shtml.

[ANL 2016b] Argonne National Laboratory (ANL). Reactors Designed by Argonne National Laboratory: Early Exploration (CP-1, CP-2, CP-3). 2016 Last Revised June 1, 2016 [cited Accessed June 2, 2016; Available from: http://www.ne.anl.gov/About/reactors/early-reactors.shtml.

[ANL 2016c] Argonne National Laboratory (ANL). Reactors Designed by Argonne National Laboratory: Integral Fast Reactor (IFR). 2016 Last Revised June 1, 2016 [cited Accessed July 7, 2016; Available from: http://www.ne.anl.gov/About/reactors/integral-fast-reactor.shtml.

[ASME 1987] American Society of Mechanical Engineers (ASME), An International Mechanical Engineering Landmark: The Vallecitos Boiling Water Reactor. 1987. p. 8 pages.

[ANS 2004] Lawson, G. and C. Krause, Documenting history: Minutes of the New Piles Committee meetings, in Nuclear News. 2004, American Nuclear Society (ANS). p. 36-38.

[Beeley N/A] Beeley, R.J. and J.E. Mahlmeister, Operating Experience with the Sodium Reactor Experiement and Its Application to the Hallam Nuclear Power Facility. Unknown year, Atomics International. p. 47 pages.

[Bettis 1957] Bettis, E., et al., The Aircraft Reactor Experiment—Operation. Nuclear Science and Engineering, 1957. 2(6): p. 841-853.

[Bereznai 2011] Bereznai, G., Introduction to CANDU Systems and Operation. 2011, University of Ontario Institute of Technology. p. 258 pages.

[BNL 1994] Higinbotham, W.A., Material Unaccounted for at the Southwest Experimental Fast Oxide Reactor. 1994, Brookhaven National Laboratory (BNL)

[Boudreau 1981] Boudreau, J., The American Breeder Reactor Program Gets a Second Chance, in Los Alamos Science. 1981.

[Boing 1991] Boing, L.E. The Experimental Boiling Water Reactor (EBWR) Decontamination and Decommissioning (D&D) Project. in Waste Management Symposium. 1991.

Brown 2009] Brown, M. Canada's Nuclear History Chronology. 2009 Last Revised November 25, 2009 [cited Accessed July 5, 2016; Available from: https://cns-snc.ca/media/history/canadian_nuclear_history.html.

[Bunker 1983] Bunker, M.E., Early Reactors: From Fermi's Water Boiler to Novel Power Prototypes, in Los Alamos Science. 1983: (URL: http://large.stanford.edu/courses/2012/ph241/tilghman2/docs/00416628.pdf). p. 8 pages.

[Burness 1980] Burness, H.S., W.D. Montgomery, and J.P. Quirk, The Turnkey Era in Nuclear Power. Land Economics, 1980. 56(2): p. 188-202.

[Cantello 2003] Cantello, G.W., The Roles Played by the Canadian General Electric Company's Atomic Power Department in Canada's Nuclear Power Program: Work, Organization and Success in APD, 1955 - 1995, in Arts and Science. 2003, Trent University: https://www.cns-snc.ca/media/history/CGE_APD_History/CGE_APD_History.html.

[CNL 2015] Canadian Nuclear Laboratories (CNL). Message from the President: The Future of NRU and CNL. 2015 Last Revised February 6, 2015 [cited Accessed July 5, 2016; Available from: http://www.cnl.ca/en/home/news-and-publications/bulletins/2015/NRU_decision.aspx.

[CNL 2016] Canadian Nuclear Laboratories (CNL). National Research Universal. 2016 Last Revised July 4, 2016 [cited Accessed July 5, 2016; Available from: http://www.cnl.ca/en/home/facilities-and-expertise/nru/default.aspx.

[Cochran 1964] Cochran, J.D. and J.E. Owens, Initial Operation of the Sodium Graphite Reactor at the Hallam Nuclear Power Facility, in American Power Conference - 26th Annual Meeting, C.P.P. District and A. International, Editors. 1964: Chicago, Illinois.

[Cochran 2009] Cochran, T.B., H.A. Feiveson, and F.v. Hippel, Fast Reactor Development in the United States. Science & Global Security, 2009. 17: p. 109-131.

[Davis 2011a] Davis, W. Elk River 1. Atomic Power Review, 2011. (URL: http://atomicpowerreview.blogspot.com/2011/09/apr-atomic-journal-elk-river-1.html).

[Davis 2011b] Davis, W. Atomic History: Dresden-1. Atomic Power Review, 2011. (URL: http://atomicpowerreview.blogspot.com/2011/06/atomic-history-dresden-1.html).

[Davis 2012] Davis, W. The MTR - Gone now, but not forgotten. ANS Nuclear Cafe, 2012. (URL: http://ansnuclearcafe.org/2012/10/25/mtr-gone-now-but-not-forgotten/).

[Davis 2013] Davis, W. The Hook-Ons. ANS Nuclear Cafe, 2013.

[Davis 2014a] W. Eisenhower's Atomic Power for Peace III: CAP and Power Demonstration Reactors. 2014. (URL: http://ansnuclearcafe.org/2014/03/20/eisenhowers-atomic-power-for-peace-iii-cap-and-power-demonstration-reactors/).

[Davis 2014b] Davis, W. They Harnessed the ATOM – the first Navy prototype nuclear plant. ANS Nuclear Cafe, 2014. (URL: http://ansnuclearcafe.org/2014/10/10/they-harnessed-the-atom-the-first-navy-prototype-nuclear-plant/).

[Davis 2014c] Davis, W. Pathfinder: A Path Not Taken. ANS Nuclear Cafe, 2014. June 13, 2014 (URL: http://ansnuclearcafe.org/2014/06/13/pathfinder/).

[Davis 2016] Davis, W. Nuclear Energy for Puerto Rico. ANS Nuclear Cafe, 2016. Accessed June 10, 2016.

[DOE 1962] Hewlett, R.G. and O.E. Anderson, The New World, 1939-1946. 1962, US Department of Energy, The Pennsylvania State University Press.

[DOE 1978] US Department of Energy (DOE), Safety evaluation report related to operation of Fast Flux Test Facility. 1978.

[DOE 1979] Huntsman, L.K. and R.H. Meservey, Sodium Removal from Hallam Reactor Components. 1979, EG&G Idaho Incorporated, US Department of Energy (DOE). p. 60 pages.

[DOE 1982a] US Department of Energy (DOE), Program Summary for the Civilian Reactor Development Program. 1982. p. 65 pages.

[DOE 1982b] Horton, K.E., Status of Fast Breeder Reactor Development in the United States of America. 1982, US Department of Energy (DOE).

[DOE 1983] Buck, A.L., A History of the Atomic Energy Commission 1983, US Department of Energy (DOE). p. 42 pages.

[DOE 1987] Horton, K.E., Status of Liquid Metal Fast Reactor Development in the United States of America. 1987, US Department of Energy (DOE). p. 17 pages.

[DOE 2011] US Department of Energy (DOE), The History of Nuclear Energy. 2011. p. 19 pages.

[DOE 2015a] US Department of Energy (DOE). 400 Area/Fast Flux Test Facility. 2015 Last Revised September 28, 2015 [cited Accessed July 7, 2016; Available from: http://www.hanford.gov/page.cfm/400areafftf.

[DOE 2015b] Flanagan, G., Previous Experience "Licensing/Authorizing" Non-LWRs in the US—How It Was Done And Who Did What. 2015, US Department of Energy (DOE). p. 10 pages.

[DOE 2016] O'Connor, T.J., Test and Demonstration Planning Study. 2016, US Department of Energy. p. 12 pages.

[EIA 2011] US Energy Information Administration (EIA). Nuclear Reactor Operational Status Tables. 2011 Last Revised November 22, 2011 [cited Accessed July 6, 2016; Available from: https://www.eia.gov/nuclear/reactors/stats_table3.html.

[EPRI 1984] Electric Power Research Institute (EPRI), Conceptual Breeder Design Completed, in EPRI Journal. 1984: (URL: http://eprijournal.com/wp-content/uploads/2016/01/1984-Journal-No.-9.pdf). p. page 43.

[Everett 1978] Everett, J.L. and E.J. Kohler, Thorium and Gas Cooled Reactors Peach bottom unit no. 1: A high performance helium cooled nuclear power plant. Annals of Nuclear Energy, 1978. 5(8): p. 321-335.

[Exelon 2015] Exelon Corporation. R.E. Ginna Nuclear Power Plant Returns to Service. 2015 Last Revised November 9, 2015 [cited Accessed July 6, 2016; Available from: http://www.exeloncorp.com/newsroom/r-e-ginna-nuclear-power-plant-returns-to-service.

[Farabee 2006] Farabee, O.A. and W.R. Church, Closure of the Fast Flux Test Facility: History, Status, Future Plans. 2006. p. 10 pages.

[Gammell 2016] Gammell, B. Connecticut Yankee and Millstone: 48 Years of Nuclear Power. 2016 Last Revised [cited Accessed July 6, 2016; Available from: http://connecticuthistory.org/connecticut-yankee-and-millstone-46-years-of-nuclear-power/.

[GIF 2002] Generation IV International Forum (GIF), A Technology Roadmap for Generation IV Nuclear Energy Systems. 2002.

[Habush 1974] Habush, A.L. and R.F. Walker, Special Issue: High Temperature Gas-Cooled Power Reactors Fort St. Vrain nuclear generating station construction and testing experience. Nuclear Engineering and Design, 1974. 26(1): p. 16-26.

[Hanford 2016] Hanford.gov. B Reactor. 2015 Last Revised November 16, 2015 [cited Accessed August 30, 2016; Available from: http://www.hanford.gov/page.cfm/BReactor.

[Haroldsen 2008] Haroldsen, R., The Story of the Borax Nuclear Reactor and the EBR-I Meltdown. 2008. p. 104 pages.

[Hewlett 1974] Hewlett, R.G. and F. Duncan, Nuclear Navy. 1974, (URL: http://energy.gov/sites/prod/files/2013/08/f2/HewlettandDuncanNuclearNavyComplete.pdf): The University of Chicago Press. 494 pages.

[Herbert 2012] Herbert, R. and G. Root, Historic American Engineering Record: Humboldt Bay Power Plant, U.S. Department of the Interior, Editor. 2012: Washington, DC.

[Hill 2013] Hill, C.N., An Atomic Empire: A Technical History of the Rise and Fall of the British Atomic Energy Programme. 2013, London, UK: Imperial College Press.

[H.R. 1414 2016] H.R. 1414 — 100th Congress: Price-Anderson Amendments Act of 1988. 1987: October 18, 2016 https://www.govtrack.us/congress/bills/100/hr1414

[IAEA 2002] International Atomic Energy Agency (IAEA), Heavy Water Reactors: Status and Projected Development. 2002. p. 170 pages.

[IAEA 2016] International Atomic Energy Agency (IAEA). Elk River. 2016 Last revised April 13, 2016 [cited Accessed April 14, 2016; Available from: https://www.iaea.org/PRIS/CountryStatistics/ReactorDetails.aspx?current=596.

[INL 1979] Reeder, D.L. and V.T. Berta, Loss-of-Fluid Test (LOFT) Facility. 1979, Idaho National Engineering Lab. p. 6 pages.

[INL 2010] McFarlane, H.F., Fast Reactor Operating Experience in the U.S. 2010, Idaho National Laboratory.

[INL 2012] Idaho National Laboratory (INL), Experimental Breeder Reactor-I. 2012. p. 2 pages.

[INL 2013] Idaho National Laboratory (INL), Transient Reactor Test Facility. 2013. p. 2 pages.

[JCAE 1952] Atomic Power and Private Enterprise, in Joint Committee on Atomic Energy (JCAE). 1952, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00002206&mediaType=application/pdf. p. 429 pages.

[JCAE 1954] Report of the Subcommittee on Research and Development on the five-year power reactor development program proposed by the Atomic Energy Commission, in Joint Committee on Atomic Energy (JCAE). 1954, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00001543&mediaType=application/pdf. p. 31 pages.

[JCAE 1955] Current Statement of the Atomic Energy Commission on the Five-Year Reactor Development Program to the Subcommittee on Research and Development, in Joint Committee on Atomic Energy (JCAE). 1955, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00001542&mediaType=application/pdf. p. 23 pages.

[JCAE 1956] Development, Growth, and State of Atomic Energy Industry: Statutory Hearings before the Joint Committee on Atomic Energy of the United States, in Joint Committee on Atomic Energy (JCAE). 1956, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00002081&mediaType=application/pdf. p. 603 pages.

[JCAE 1957a] Development, Growth, and State of the Atomic Energy Industry: Hearings before the Joint Committee on Atomic Energy of the United States, in Joint Committee on Atomic Energy (JCAE). 1957, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00002577&mediaType=application/pdf. p. 866 pages.

[JCAE 1957b] Review of Proposals Under Power Demonstration Program: Hearing before the Subcommittee on Legislation of the Joint Committee on Atomic Energy, in Joint Committee on Atomic Energy (JCAE). 1957, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00001885&mediaType=application/pdf. p. 67 pages.

[JCAE 1963] Hearings before the Subcommittee on Legislation of the Joint Committee on Atomic Energy, in Joint Committee on Atomic Energy (JCAE). 1964, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00002072&mediaType=application/pdf p. 262 pages.

[JCAE 1965] Development, Growth, and State of Atomic Energy Industry: Hearings before the Joint Committee on Atomic Energy of the United States, in Joint Committee on Atomic Energy (JCAE). 1965, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00002235&mediaType=application/pdf. p. 627 pages.

[JCAE 1966] Enrico Fermi Reactor: Use for Irradiation Testing and Background Information: Hearings before the Joint Committee on Atomic Energy of the United States, in Joint Committee on Atomic Energy (JCAE). 1966, Stanford University: (URL: http://sul-derivatives.stanford.edu/derivative?CSNID=00000572&mediaType=application/pdf). p. 276 pages.

[JCAE 1974] Developments in the Energy Industry in General and the Nuclear Power Industry in Particular, Pursuant to Section 202 of Atomic Energy Act of 1954, as Amended: Hearings before the Joint Committee on Atomic Energy of the United States, in Joint Committee on Atomic Energy (JCAE). 1974, Stanford University: http://sul-derivatives.stanford.edu/derivative?CSNID=00000639&mediaType=application/pdf. p. 612 pages.

[Johnson 2014] Johnson, D., Lessons from the Licensing of Fort Saint Vrain, in 3rd Annual EPRI Nuclear Fuel Cycle Assessment Workshop, ABS Group, Editor. 2014: Nashville, TN.

[Kittel 1958] Kittel, J.H., M. Novick, and R.F. Buchanan, The EBR-I Meltdown—Physical and Metallurgical Changes in the Core. Nuclear Science and Engineering, 1958. 4(2): p. 180-199.

[Lipa 2012] Lipa, C. The Challenge of Decommissioning a One-of-a-Kind Reactor. 2012 Last Revised June 8, 2012 [cited Accessed July 6, 2016; Available from: https://public-blog.nrc-gateway.gov/2012/06/08/the-challenge-of-decommissioning-a-one-of-a-kind-reactor/.

[LANL 2004] Harvey, D.W., et al., Historical Context of the Omega Reactor Facility, Technical Area 2 2004, Los Alamos Natioanl Laboratory (LANL). p. 120 pages.

[Lovering 2016] Lovering, J.R., A. Yip, and T. Nordhaus, Historical construction costs of global nuclear power reactors. Energy Policy, 2016. 91: p. 371-382.

[MacLeod 2015] MacLeod, I. World's oldest operating reactor, in Chalk River, to close in 2018. 2015.

[Martin 2004] Martin, D., Ontario's Nuclear Generating Facilities: A History and Estimate of Unit Lifetimes and Refurbishment Costs. 2004, Pembina Institute, Canadian Environmental Law Association: Toronto.

[Mazuzan 1984] Mazuzan, G.T. and S. Walker, Controlling the Atom – The Beginning of Nuclear Regulation, 1946 – 1962. Vol. (URL: http://www.ucpress.edu/op.php?isbn=9780520051829). 1984: Berkeley: University of California Press.

[McConnell 2002] McConnell, L. NPD: Canada's First Nuclear Power Station in Special seminar hosted by the Chalk River Branch of the Canadian Nuclear Society. 2002. Deep River, Ontario: Canadian Nuclear Society (CNS).

[Morrissey 1979a] Morrissey, J.E., et al., General Electric Test Reactor NTD Silicon Development Program, in Neutron Transmutation Doping in Semiconductors, J.M. Meese, Editor. 1979, Springer US: Boston, MA. p. 171-179.

[Morrissey 1979b] Morrissey, J., Commercial fuel testing in the GETR (General Electric Test Reactor). Nuclear Engineering International, 1979. 24(286): p. 48-53.

[Morris 1993] Morris, A.L., A History of Carolina Power and Light Company 1958 - 1992 1993. p. 9 pages.

[NARA 2016] US National Archives and Records Administration (NARA). Records of the Rural Electrification Administration. 2016 [cited Accessed June 10, 2016; Available from: http://www.archives.gov/research/guide-fed-records/groups/221.html.

[NEI 2016] Nuclear Energy Institute (NEI). US Nuclear Power Uprates by Plant (1977-2016). 2016 Last Revised May 2016 [cited Accessed July 6, 2016.

[Nielsen 2004] Nielsen, D.L., Fast Flux Test Facility - A History of Safety and Operational Excellence. 2004, Fluor Hanford. p. 30 pages.

[NJ Department of Planning 2014] NJ Department of Planning, E., and Parks & Recreation, Welcome to Oyster Creek Generating Station. 2014.

[NRC 1979] US Nuclear Regulatory Commission (NRC), Safety Evaluation Report related to Operation of Fast Flux Test Facility Supplement 1. 1979. p. 152 pages.

[NRC 2004] US Nuclear Regulatory Commission (NRC), Fort Saint Vrain Gas Cooled Reactor Operational Experience. 2004.

[NRC 2014a] US Nuclear Regulatory Commission (NRC). History 2014 Last revised October 2, 2014 [cited Accessed January 7, 2016; Available from: http://www.nrc.gov/about-nrc/emerg-preparedness/history.html.

[NRC 2014b] US Nuclear Regulatory Commission (NRC). Yankee Rowe. 2014 Last Revised May 30, 2014 [cited Accessed June 9, 2016; Available from: http://www.nrc.gov/info-finder/decommissioning/power-reactor/yankee-rowe.html.

[NRC 2015a] US Nuclear Regulatory Commission (NRC). Indian Point - Unit 1 2015 Last Revised June 5, 2015 [cited Accessed June 3, 2016; Available from: http://www.nrc.gov/info-finder/decommissioning/power-reactor/indian-point-unit-1.html.

[NRC 2015b] US Nuclear Regulatory Commission (NRC). La Crosse Boiling Water Reactor 2015 Last Revised June 12, 2015 [cited Accessed June 10, 2016; Available from: http://www.nrc.gov/info-finder/decommissioning/power-reactor/lacrosse-boiling-water-reactor.html.

[NRC 2016a] US Nuclear Regulatory Commission (NRC). San Onofre - Unit 1. 2016 Last Revised April 19, 2016 [cited Accessed June 9, 2016; Available from: http://www.nrc.gov/info-finder/decommissioning/power-reactor/san-onofre-unit-1.html.

[NRC 2016b] US Nuclear Regulatory Commission (NRC). R. E. Ginna Nuclear Power Plant 2016 Last Revised April 4, 2016 [cited Accessed July 6, 2016; Available from: http://www.nrc.gov/info-finder/reactors/ginn.html.

[NRC 2016c] US Nuclear Regulatory Commission (NRC). General Electric Co. - General Electric Test Reactor (GETR). 2016 Last Revised June 6, 2016 [cited Accessed June 27, 2016; Available from: http://www.nrc.gov/info-finder/decommissioning/research-test/general-electric-co-getr.html.

[NRC 2016d] US Nuclear Regulatory Commission (NRC). Dresden - Unit 1 2016 Revised April 12, 2016 [cited Accessed June 9, 2016; Available from: http://www.nrc.gov/info-finder/decommissioning/power-reactor/dresden-nuclear-power-station-unit-1.html.

[NRC 2016e] US Nuclear Regulatory Commission (NRC). Big Rock Point. 2015 Last revised April 01, 2015 [cited Accessed June 4, 2016; Available from: http://www.nrc.gov/info-finder/decommissioning/power-reactor/big-rock-point.html.

[NRC 2016f] US Nuclear Regulatory Commission (NRC). Fermi, Unit 1 2015 Last revised November 2, 2015 [cited Accessed January 7, 2016; Available from: http://www.nrc.gov/info-finder/decommissioning/power-reactor/enrico-fermi-atomic-power-plant-unit-1.html.

[NSF 1963] Allen, W., Nuclear Reactors for Generating Electricity: U.S. Development from 1946 to 1963. 1977, Rand Corporation, National Science Foundation (NSF). p. 101 pages.

[NSF 1977a] Arturo Gandara, Electric Utility Decisionmaking and the Nuclear Option. 1977, Rand Corporation, National Science Foundation (NSF): Santa Monica, CA. p. 120 pages.

[NSF 1977b] Perry R., et al., Development and Commercialization of the Light Water Reactor, 1946-1976. 1977, Rand Corporation, National Science Foundation (NSF): Santa Monica, CA. p. 130 pages.

[ORNL 1952] Buck, J.H. and C.F. Leyse, Materials Testing Reactor Project Handbook. 1952, Argonne National Laboratory, Oak Ridge National Laboratory. p. 286 pages.

[ORNL 1957] Briggs, R.B., et al., Homogeneous Reactor Project: Quarterly Progress Report for Period Ending July 31, 1957. 1957, Oak Ridge National Laboratory. p. 195 pages.

[ORNL 1958] Oak Ridge National Laboratory (ORNL), The ORNL Gas-Cooled Reactor. 1958. p. 22 pages.

[ORNL 1963] Blosser, T.V. and R.M. Freestone, A Radiation Study of the Westinghouse Testing Reactor. 1963, Oak Ridge National Laboratory (ORNL). p. 30 pages.

[ORNL 1973] Heddleson, F.A., Design Data and Safety Features of Commercial Nuclear Power Plants. 1973, Oak Ridge National Laboratory (ORNL). p. 250 pages.

[ORNL 1976] Heddleson, F.A., Design Data and Safety Features of Commercial Nuclear Power Plants. 1976, Oak Ridge National Laboratory (ORNL). p. 184 pages.

[ORNL 2001] Greene, S.R., Molten Salt Reactors: Technology History, Status, and Promise. 2001, Oak Ridge National Laboratory (ORNL).

[ORNL 2004] Copinger, D.A. and D.L. Moses, Fort Saint Vrain Gas Cooled Reactor Operational Experience 2004, Oak Ridge National Laboratory (ORNL), US Nuclear Regulatory Commission (NRC). p. 62 pages.

[ORNL 2009] Rosenthal, M.W., An Account of Oak Ridge National Laboratory's Thirteen Nuclear Reactors 2009, Oak Ridge National Laboratory. p. 80 pages.

[Oulahan 1959] Oulahan, C., Part IV Federal Statutory and Adminstrative Limitations upon Atomic Activities, in Atoms and the Law. 1959, University of Michigan Law School. p. 152 pages.

[Planchon 1987] Planchon, H.P., et al., Results and Implications of the EBR-II Inherent Safety Demonstration Tests, in International Topical Meeting on Advances in Reactor Physics, Mathematics, and Computation. 1987: Paris, France. p. 22 pages.

[PNNL 2011] Mcdowell, B., et al., High Temperature Gas Reactors: Assessment of Applicable Codes and Standards. 2011, Pacific Northwest National Laboratory (PNNL). p. 41 pages.

[Ragheb 2015] Ragheb, M., Sodium-Graphite Reactor Experiement, SRE Incident. 2015.

[RNL 1996] Nonbol, E., Description of the Advanced Gas Cooled Type of Reactor (AGR). 1996, Riso National Laboratory (RNL). p. 88 pages.

[RNL 1998] Jensen, S.E. and E. Nonbol, Description of the Magnox Type of Gas Cooled Reactor. 1998, Riso National Laboratory. p. 83 pages.

[Rosenthal 1968] Rosenthal, M., P. Kasten, and R. Briggs, Molten-Salt Reactors - History, Status, and Potential. Nuclear Applications and Technology, 1969. 8(2): p. 107-117.

[Stason 1959] Stason, E.B., S.D. Estep, and W.J. Pierce, Atoms and the Law. 1959, University of Michigan Law School.

[Sneddon 2010] Sneddon, S.T., An Investigation of British Nuclear Policy 1945-2005. 2010, University of Northampton: http://nectar.northampton.ac.uk/3551/1/Sneddon20103551.pdf. p. 214 pages.

[Serp 2014] Serp, J., et al., The molten salt reactor (MSR) in generation IV: Overview and perspectives. Progress in Nuclear Energy, 2014. 77(0): p. 308-319.

[Stoffel 2000] Stoffel, J., Dreams of Flight: The NEPA and ANP Programs. 2000, University of Wisconsin- Madison. p. 23 pages.

[Taylor 2016] Taylor, S., The Fall and Rise of Nuclear Power in Britain: A History. 2016, Cambridge, UK: UIT Cambridge Ltd.

[Tompkins 2006] Tompkins, B. Big Rock Point: From groundbreaking to greenfield. 2006. (URL: http://www2.ans.org/pubs/magazines/nn/docs/2006-11-3.pdf), 8 pages.

[Wald 2010] Wald, M.L. Oyster Creek Reactor to Close by 2019. New York Times, 2010.

[Walters 1999] Walters, L.C., Thirty years of fuels and materials information from EBR-II. Journal of Nuclear Materials, 1999. 270: p. 39-48.

[Wakeford 2007] Wakeford, R., The Windscale reactor accident—50 years on. Journal of Radiological Protection, 2007. 27(3): p. 211.

[Westfall 2004] Westfall, C., Vision and reality: The EBR-II story, in Nuclear News. 2004: (URL: http://www.ne.anl.gov/About/reactors/EBR2-NN-2004-2-2.pdf).

[Widner 2009] Widner, T., Reactor Development and Operations at Los Alamos. 2009. p. 66 pages.

[Whitlock 2005] Whitlock, J.J., The Historical Significance of the Douglas Point Nuclear Power Plant, C.N. Society, Editor. 2005. p. 4 pages.

[Whitlock 2016a] Whitlock, J.J., Pickering - A Nuclear Generating Station, C.N. Society, Editor. 2016. p. 3 pages.

[Whitlock 2016b] Whitlock, J.J., Nuclear Power Demonstration (NPD) Backgrounder, C.N. Society, Editor. 2016. p. 2 pages.

[Winfield 2006] Winfield, M., et al., Nuclear Power in Canada: An Examination of Risks, Impacts and Sustainability. 2006, The Pembina Institute. p. 130 pages.

[WNA 2015] World Nuclear Association (WNA). Nuclear Power in Canada. 2015 Last Revised December 15, 2015 [cited Accessed January 8, 2016; Available from: http://www.world-nuclear.org/info/Country-Profiles/Countries-A-F/Canada--Nuclear-Power/.

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