Report to Congress on

Small Modular Nuclear Reactors

May 2001

U.S. Department of Energy
Office of Nuclear Energy, Science and Technology

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William D. Magwood, IV
Director, Office of Nuclear Energy,
Science and Technology

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LIST OF ACRONYMS

AEC	U.S. Atomic Energy Commission
ANPP	Army Nuclear Power Program
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CNEA	Argentinean National Atomic Energy Commission
CRIEPI	Central Research Institute of Electric Power Industry (Japan)
DBA	Design Basis Accident
DOE	U.S. Department of Energy
EPZ	emergency planning zone
GDC	General Design Criteria
HTGR	High Temperature Gas-cooled Reactor
IAEA	International Atomic Energy Agency
IRIS	International Reactor Innovative and Secure
JAERI	Japanese Atomic Energy Research Institute
kWe	kilowatt-electric
kWh	kilowatt-hour
LEU	low enriched uranium
LMR	liquid metal-cooled reactor
LOCA	Loss of Coolant Accident
LOFA	Loss of Flow Accident
LOFC	Loss of Forced Circulation
LWR	light water reactor
MH-1A	Mobile High Power Nuclear Power Plant
MMR	Multi-Module Reactor
MWe	megawatt-electric
MWt	megawatt-thermal
NERI	Nuclear Energy Research Initiative
NRC	Nuclear Regulatory Commission
OKBM	Special Design Bureau for Mechanical Engineering (Russia)
PAG	Protective Action Guide
PRA	Probabilistic Risk Assessment
PRDP	Power Reactor Development Program
PTS	Pressurized Thermal Shock
PWR	pressurized water reactor
SBO	Station Blackout
SM-1	Stationary Medium Power Prototype Number1
SMR	Small Modular Reactor

SSC	systems, structures, and components
SSR	Solid-State Reactor

EXECUTIVE SUMMARY

This report examines the feasibility of, and issues associated with, the deployment of small nuclear power plants as a potential option in providing electric power to remote areas that are deficient in transmission and distribution infrastructures. Remote communities pose special challenges in providing electric power because it is likely that there will be a shortage of trained personnel, higher expense, difficulty in shipping and storing fuel, and power requirements that are relatively small and variable. A power generating system serving such areas must be very reliable, as remoteness implies restricted accessibility for repair crews. The conclusions of the study offer sufficient reason for optimism that the most technically mature small modular reactor (SMR) designs and concepts have the potential to be economical and could be made available for deployment before the end of the decade, provided that certain technical and licensing issues are addressed.

Based on the current power usage of typical remote communities in the United States, a maximum electricity generating capacity of 50 megawatts-electric (MWe) was set for selecting the SMR designs and concepts to be studied. Plant characteristics were evaluated on the basis of their ability to satisfy the criteria that were specified in the Senate Report 106-395 accompanying the Fiscal Year 2001 Energy and Water Development Appropriation: inherent safety, cost-effectiveness, resistance to sabotage and diversion of nuclear materials, infrequent refueling, the level of factory fabrication, and transportability to remote sites. This report describes the results of the study, dividing the discussion into three critical aspects: technical, regulatory, and economic issues.

Both existing SMR designs and proposed SMR concepts from domestic and foreign sources were reviewed. Emphasis was placed on assessing the technical viability of each concept or design, including identifying any innovative features and highlighting any further research that might be required for successful development. Unconventional features of SMRs which have licensing implications, including how issues might be resolved within the current Nuclear Regulatory Commission (NRC) licensing and regulatory environment, were addressed. Expected costs for SMRs have been estimated and compared with current electricity generation prices. Before these new designs and concepts can be deployed, however, vendors must recognize that effective communication with the public will be as important as the development of their respective designs and concepts.

Technical Assessment:
To simplify the designs and to reduce or eliminate potential accident consequences, the SMRs all make greater use of inherent safety features than do existing larger commercial plants. For example, inherent safety may be achieved through fuel designs that are able to withstand extreme temperatures without loss of the fuel’s integrity. Almost all of the designs and concepts rely on natural circulation of the coolant in emergency modes and many SMRs additionally rely on natural circulation for cooling of the core during normal operation.

Since most of the SMRs studied use small inventories of low enriched uranium (LEU)-based fuels (defined as less than 20 percent U^{235} content of the total uranium), the power plants would not be appealing targets for sabotage or diversion. Diversion resistance of the spent fuel is supported by the accumulation of highly radioactive fission products formed during reactor operation.

The refueling intervals range between 1 and 15 years for the SMRs reviewed in this study. Spent-fuel storage requirements vary, with some transportable SMRs being refueled at maintenance centers, that adds an additional level of diversion resistance and spent fuel storage, while other SMRs follow the usual practice of on-site refueling, that may provide operational advantages. The degree of factory fabrication and modular construction varies greatly between the designs and concepts reviewed in this study. However, factory fabrication of a power plant in modules results in shorter construction time, ease of transportability, and simpler on-site assembly in remote locations.

Licensing and Regulatory Issues:
The current NRC regulatory framework for ensuring plant safety has three main elements: reactor safety, radiation safety, and plant security. Because many SMRs use different approaches to satisfy these areas of safety, including inherent safety characteristics, a more simplified licensing and regulatory process than that used for light water reactors (LWR) would be appropriate. The potential impact on a SMR regulatory framework is presented, using some typical advanced gas-cooled SMR designs. Potential licensing issues that may be encountered, such as non-traditional containment concepts, are highlighted. Issues that could affect some SMRs' regulatory approval include pyrocarbon coated particle fuel performance and reactor containment design.

Economic Competitiveness:
The economic competitiveness of a generic 50 MWe and 10 MWe SMRs is estimated and compared with current generation costs of electricity in selected remote locations. For this comparison, the delivered cost of electricity charged to industrial customers by selected utilities in Alaska and Hawaii is used as a baseline for remote or isolated communities. For a generic 50 MWe SMR, the range of electricity cost is estimated at 5.4 to 10.7 cents per kilowatt-hour (kWh). The range of cost for a 10 MWe SMR is 10.4 to 24.3 cents per kWh. Since the industrial rate for electricity charged by selected Alaska and Hawaii utilities varied from 5.9 to 36.0 cents per kWh, (depending on the location, type and size of the power plant, fuel cost, and ease of transporting fuel), SMRs could be a competitive option.

Conclusion:
The study found no substantive technical issues to hinder development and deployment of SMRs, and initial estimates of the electricity generation costs are comparable to, if not better than, those for current electricity supplies in typical remote areas. Furthermore, research into nuclear reactor technology and small power reactors by the Army has shown that nuclear power facilities can be safely constructed and operated in remote areas. However, some of the more viable SMRs, such as the Remote-Site Modular Helium Reactor (RS-MHR), involve licensing issues that are outside of the traditional NRC

light water reactor experience (as some SMRs are cooled by gas or liquid metals) and therefore, the current regulatory environment may not be entirely applicable. Also, it would be beneficial to further refine the cost estimates in order to provide an adequate basis to make any decisions regarding development and deployment of SMRs for use in remote communities.

In conclusion, the Department believes there are SMR designs and concepts being developed that have potential for deployment in remote communities.

1 INTRODUCTION

Recent trends in nuclear reactor power plant designs have led to the development of numerous designs and proposed concepts for small modular nuclear reactors with a wide variety of safety and engineering features. These new power plants may prove attractive for remote communities where the current ability to generate electric power depends on costly shipments of fuels, resulting in relatively expensive and possibly uncertain electricity supplies. In fiscal year 2001, Congress provided funding and direction to perform a study on this issue, according the following language contained in the Senate Report (S. Rept.) 106-395:

> The Committee is aware that recent improvements in reactor design might make feasible small modular reactors with attractive characteristics for remote communities that otherwise must rely on shipments of relatively expensive and sometimes environmentally undesirable fuels for their electric power. To be acceptable, such a reactor would have to be inherently safe, be relatively cost effective, have intrinsic design features which would deter sabotage or efforts to divert nuclear materials, have infrequent refueling, and be largely factory constructed and deliverable to remote sites. The Committee recommendation provides $1,000,000 for the Department to undertake a study to determine the feasibility of and issues associated with the deployment of such small reactors and provide a report to Congress by May 2001.

This report describes the results of the study performed in response to this direction. The study specifically addresses the characteristics mentioned in S. Rept. 106-395, dividing the discussion into three critical aspects: technical, regulatory, and economic issues. Information on numerous designs and concepts was gathered from submissions received in response to the announcement of this study in *Commerce Business Daily*, February 26, 2001, and the *Federal Register*, February 27, 2001, as well as from literature and Internet searches. The depth of information received, however, varied. For instance, more detailed information on the Remote-Site Modular Helium Reactor, was received in response to the announcement for information on small modular reactors (SMR) and is reflected in the length of its review in Chapter 3. For most of the other SMR designs and concepts reviewed, the basic layout of the facility has been established, perhaps even to some of the smallest details, but design tradeoffs and optimization have not been performed, and the detailed engineering to build the plant has not been completed. As a result, this report contains no illustrations of any plant designs or concepts.

For the technical assessment, emphasis has been placed on evaluating the viability and maturity of each SMR, including identification of any recent or innovative developments, and the time that would be required for further development and potential deployment. The effect of current licensing requirements has been explored, and possible beneficial modifications have been identified. Estimates of the expected costs of generic SMRs and comparisons with current electricity rates in typical remote locations have been provided.

1.1 Determination of Reactor Size for Remote Siting

There can be a wide variation in what would be considered a “small” reactor. For many utilities in the continental United States, anything less than 300 megawatt-electric (MWe) would be considered small, based on the reactors currently in operation. For utilities in remote areas where the population is small and plants cannot be economically connected to a power grid; the range of power plant sizes of interest is much different. The requirements that the reactor must be attractive to, and deliverable to, remote sites also imply a low-power level. To determine the needs of typical remote communities, this study reviewed the current power generating capacity of power plants in isolated locations, including islands. The upper limit for power generating capacity for the small reactors considered in this study was 50 MWe. Currently, the non-nuclear capacity that is serving these remote communities is typically in the range of 1 MWe to 20 MWe. Examples of locations where SMRs could be considered for substitute power are discussed below:

- **Alaska**
  Alaska is a large state with a widely scattered population. Aside from a few major population centers, such as Fairbanks, Anchorage, and Juneau, the remainder of Alaska’s population resides in numerous small towns that are spread throughout the state. In general, each of these towns is responsible for its own electric power or is a member of a small, regional power system. The Department of Energy (DOE) *Inventory of Power Plants* lists about 620 individual generating units in Alaska. About half of the existing power generation units are diesel engines, providing less than 1 MWe of output each. There are also many power plants in the range of 20 MWe to 40 MWe, while the five largest power plants are in the range of 89 MWe to 335 MWe. In considering possible replacement power plants, it appears that units less than 50 MWe would represent the majority of Alaskan generating capability, with units of 10 MWe or less being the most widely applicable.

- **Hawaii**
  The state of Hawaii consists of a number of islands served by six physically separate electric power systems with no interconnections. For each of these systems, the great majority of electric generating units are small in size, from 1 MWe (or smaller) to several tens of megawatts. The five largest units range between 174 MWe and 582 MWe. At this time, about three-quarters of the electricity in Hawaii is generated from petroleum-based fuels, of which 71 percent is foreign import and 29 percent is primarily from Alaska. Another 16 percent of the electric generating capacity is fueled by coal imported from Indonesia and Australia. The remaining eight percent of the states electric power is supplied by various sources, such as geothermal, hydroelectric, municipal waste and industrial overcapacity. As with Alaska, the majority of power plants are units of less than 50 MWe output, with units of 10 MWe or less being the most widely applicable.

- **International**
  There is also a large need for smaller power units internationally. Many of the conditions faced in Alaska and Hawaii exist around the world. Examples can be found in the Siberian region of Russia, which is similar to Alaska, while the small islands in Japan and other island nations have conditions that are similar to Hawaii. For many nations, additional challenges include the lack of any reliable electricity grid, requiring power to be generated more locally, even for larger population centers. Since electricity demand per capita is presently very low in many of these nations, only smaller power plants would be useful. Based on design work being performed in other countries on small nuclear power plants, the power output range of interest appears to be 20 MWe to 50 MWe.

1.2 Definition of Modular Reactors

Historically, the term "modular," when applied to nuclear reactors, has designated concepts where large power plant complexes were constructed from clusters of smaller reactors, with each reactor being described as a "module." For example, this is the current use of the term in the 110 MWe Pebble Bed Modular Reactor (PBMR) design. Such reactor designs are of interest for many reasons, including the ability to add generating capacity incrementally to a large electricity grid, ease of transport and construction, and sharing of overhead by controlling more than one reactor from a single control room. However, for small reactors the term "modular" can also describe a single reactor that is assembled from factory-fabricated modules, where each module represents a portion of the finished plant. Even though current large nuclear power plants incorporate factory-fabricated components, a substantial amount of field work is required to assemble the components into an operational power plant. The use of modules implies that assembly has been reduced to limited activities such as connecting the modules, greatly reducing the amount of field work required, and simplifying completion. Maintenance may also be simplified, since defective modules can be removed and replaced, with repairs made afterward at a central facility. Taking this approach, this use of modules increases the ability to deploy a reactor in remote locations. As a consequence, in this study, these types of "modular" design features were considered to be very important.

1.3 Definition of Inherent Safety

While all current U.S. nuclear reactors are safe, the concept of inherent safety can be considered to supplement or replace the need for traditional safety systems. The traditional nuclear reactor safety systems have usually been referred to as "active engineered safety systems," since they involve engineered components that are required to perform some action in response to reactor conditions or operator commands. Even though many such active systems can partly rely on natural physical phenomena, such as gravity, they also involve electrical or mechanical operations that have finite probabilities, albeit small, of failure. The excellent designs used today have reduced the possibility of failure of such devices to extremely low levels.

There are also engineered safety systems that operate passively, that is, there is no automatic or manual activation signal required to have the safety system perform its function. Examples of such devices are pressure relief valves, that are designed to open when pressure exceeds the force of a spring. By removing the need for actuating signals, the possibility that accidents would occur or that accident consequences would be severe is reduced. There can, however, still be malfunctions and component breakage. As a consequence, the functioning of the device may not be absolutely guaranteed, and the probability of failure can be drastically reduced to acceptable levels through the use of parallel, redundant systems.

Inherent safety is fundamentally different. In general, the use of inherent safety features assures that the reactor response to any upset condition is not determined by the functioning of engineered components, but is controlled instead by basic, inherent, physical phenomena, such as the expansion of metals with increasing temperature, or the use of buoyancy to provide flow and cooling by natural circulation. For the purposes of this report, inherent safety is viewed as a condition that is achieved without the operation or functioning of any device that is susceptible to failure. Depending on the design, it may be possible to use inherent safety features to reduce or eliminate altogether the need for active engineered safety systems. The use of inherent safety principles should further reduce or eliminate the likelihood and consequences of a reactor accident, and make reactor design and operation simpler. The use of inherent safety features is considered very important, since any benefits to operation and accident mitigation would be more significant in remote areas. In general, the SMRs studied, incorporate the best possible uses of inherent safety features. With the elimination of redundant safety systems, inherent safety also provides an economic advantage.

1.4 Resistance to Diversion of Nuclear Materials

The concept of diversion resistance is used to indicate the degree to which the reactor design or concept contains and protects nuclear materials of military significance, including that produced during operations. There are a number of interrelated considerations to be addressed when assessing the diversion resistance of a reactor design or concept, including:

- **Nuclear Materials Characteristics**
  Diversion resistance is affected by the type of nuclear materials used to fuel the reactor. Nuclear fuel is normally not highly radioactive prior to use in a reactor. The low level of radiation in unirradiated fuel results in low personnel exposure to radiation during routine handling of the fuel, especially during the manufacturing process. Since the unirradiated fuel can be easily handled, it is also more vulnerable to diversion or theft. But, if the unirradiated fuel is not useful for military purposes, diversion resistance is not compromised. There are two main ways to achieve this goal. First, it is beneficial to specify low enriched uranium (defined as less than 20 percent U^{235} content of the total uranium) for the reactor fuel, since these materials are not useful for weapons. It is also preferred that only uranium be in the unirradiated fuel, with no measurable amount of plutonium, as

plutonium could be chemically separated from unirradiated fuel with the proper equipment and knowledge.

- **Nuclear Materials Inventory**
  The low power output of small reactors tends to require only small inventories of nuclear materials. These small inventories help to discourage diversion as it would be a difficult and time-consuming effort to amass a useful amount of weapons-usable materials from such inventory levels. A small inventory also makes it easier to detect any significant diversion, since a larger fraction of the total materials inventory would have to be involved.

- **Refueling Interval**
  The refueling interval affects proliferation and diversion resistance, since each refueling offers opportunities to physically handle and remove nuclear materials. A long refueling interval provides greater assurance that the nuclear materials are in their intended locations. For the purposes of this study, a long refueling interval is also desirable from an operational standpoint. For reference, a long refueling interval would be on the order of five to ten years.

- **Capability to Produce Weapons-Grade Nuclear Materials**
  For military purposes, the usefulness of the irradiated, or spent, fuel depends on a number of factors. Probably most important are the relative isotopic concentrations of the chemical elements of interest, such as plutonium. In principle, all reactors are capable of producing nuclear materials that can be used for military purposes. It is easier, however, to produce such materials with some types of reactors than with others. For some reactors, very short irradiation periods in the reactor are sufficient to produce materials of military interest. In other reactors, such materials might be produced over an extended time period, reducing the threat. However, the irradiated fuel is more self-protecting as a result of the formation of highly radioactive fission products, that would create the need for special facilities to remove nuclear materials for weapon use.

- **Physical Form of the Reactor Fuel**
  The form of the reactor fuel can also provide diversion resistance. The pertinent issues are the size of the reactor fuel, the manner in which the reactor fuel is contained, and the distribution of the nuclear materials in the reactor fuel. For example, if the unit of assembled reactor fuel is relatively small, it would be easier to divert the materials than it would be with larger fuel assemblies. It may also be easier to disassemble certain types of fuel to recover the nuclear materials, while other forms may be more difficult. Depending on the design, the reactor fuel may either be concentrated within the core or dispersed throughout the core materials, with dispersed fuel presenting more difficulties for the recovery of nuclear materials.

2 BRIEF HISTORY OF EARLIER SMALL REACTORS

In the 1950s, the United States Army and Navy initiated research programs which resulted in the design and testing of various types of small nuclear reactors. The Army Nuclear Power Program (ANPP), which is most germane to this study, focused on reactor systems for the production of electricity in remote areas. These included a mobile reactor small enough to allow transportation by tractor trailer without dismantlement, and the placement of a reactor and electrical power generation system on a barge that could be towed where needed. The Navy nuclear program centered on the development and deployment of a reactor for operating ship systems.

2.1 Army Programs for Small Reactor Development

In 1957, the ANPP began with the development of a small, pressurized water reactor (PWR), the Stationary Medium Power Prototype Number 1 (SM-1), at Fort Belvoir, Virginia. Although SM-1 was a stationary reactor, built and operated at the base, the small power plant provided reliable power to this base and provided training for the operators of the ANPP. The SM-1 operated for 16 years, before being deactivated in 1973. Another such reactor was built and operated at Fort Greely in Alaska, and its successful operation continue after surviving the March 1964 Alaskan earthquake.

The Army experimented with using small reactors to power heavy overland cargo haulers and as substitute power in remote areas. By the late 1960s, the Army program developed a PWR that could be moved into remote areas, or locations that required supplemental electrical power. This resulted in the deployment of the Mobile High Power Nuclear Power Plant (MH-1A). The MH-1A was a single loop PWR that was built and housed on a converted ship where the engine room was removed to make the ship into a barge. This power plant provided power to the Panama Canal Zone from 1968 to 1976.

Many of the Army's problems were related to logistics. The small number of units made reactor fuel and components very expensive. High operating costs were compounded by the large number of personnel required to maintain the plant documentation. In addition, operation and maintenance costs were high because of the need to have skilled personnel available around the clock in remote locations. The Army's program, however, proved that reactors could be sited, constructed or assembled in remote places, and safely operated by properly trained personnel.

Eventually, the costs of developing and producing small mobile nuclear power plants became so expensive that the Army decided that it could no longer justify the continuation of the program. Since that time, the Army's participation in nuclear power plant research and development declined steadily and eventually stopped.

The Army experience with small nuclear reactors provided insights for more recent plant designs:

- highly skilled local labor with regard to construction scheduling and project costs is needed;
- many reactor components should be prefabricated because of short construction seasons at many remote sites; and
- reactor design standardization reduces training time, material, fabrication, maintenance, and component inventory costs.

2.2 Atomic Energy Commission Power Reactor Development Program

Also in the 1950s, the U.S. Atomic Energy Commission (AEC) authorized a program to support private industry's research and development of nuclear reactors. The AEC's Power Reactor Development Program (PRDP) initially was designed to research basic reactor types that included: pressurized water, boiling water, homogeneous core, fast breeder, and sodium graphite reactors. In addition, the program funded the construction and operation of a few small U.S. commercial reactors. The program later expanded to the research of many other reactor types.

The commercial reactors within the PRDP generally produced less than 200 MWe and provided research capability for testing and refining reactor fuels, environmental control systems, and emergency systems. The first commercial reactor built in the United States, Shippingport Atomic Power Station, a 60 MWe PWR, was constructed in less than three years, and operated successfully from December 1957 to October 1982. The Yankee Atomic Power Station, a 167 MWe PWR, was built in less than five years and operated from November 1960 until October 1991. The Peach Bottom Unit 1 reactor, a 40 MWe High Temperature Gas-cooled Reactor (HTGR), was built in four years and operated successfully for more than eight years. Big Rock Point, a 67 MWe boiling water reactor, was built in less than three years, operated successfully for 35 years from December 1962 to August 1997.

These facilities provided the engineering data that supported the vision that nuclear power could be used safely to supply electricity to the country. These plants performed well but were shut down after their proofs of concepts were complete, or when their engineered lifetimes had been met, or when they could no longer compete economically with larger more efficient power plants.

The results obtained from the PRDP's expanded research program of other reactor designs were mixed. The Piqua Nuclear Power Facility was an organic-cooled 11 MWe unit that was built and operated for demonstration purposes. The facility was shut down after less than three years of operation due to numerous problems. The Saxton Nuclear Experimental Facility, a 3 MWe unit that was part of the Experimental Power Reactor Program, was constructed in less than one year and was connected to the electrical grid. It was used as a research and training reactor to investigate improvements on PWR design, rod cluster control, and chemical shim reactivity control. This reactor operated from 1962 to 1972.

In the United Kingdom (UK), gas-cooled reactors using carbon dioxide for the reactor coolant were developed. The early UK gas reactors all produced around 50 MWe. It is interesting to note that many of these older, smaller units in England are still in operation today. Newer designs raised the output power in about 100 MWe increments, so that today, the modern Advanced Gas Reactors operating in the UK are capable of producing more than 600 MWe.

2.3 Summary

The initial research into nuclear reactor technology and small power reactors by the Army has shown that nuclear power facilities can be safely constructed and operated in remote areas. This experience provides evidence that today's remote siting concerns can be safely addressed. The experience of the AEC's PRDP demonstrated that new reactor designs of small size could be constructed, tested, and placed on the electric power grid in a relatively short time frame (e.g., less than four years), but this was prior to modern regulatory and siting requirements. The UK experience of applying a standardized gas reactor technology shows that gas reactor designs can be expanded to fit the electrical consumption needs and provide a stable source of nuclear power in today's economic environment.

3 TECHNICAL CHARACTERISTICS OF RECENT DESIGNS

In general, there has been a trend toward reactor concepts which are of standardized design and incorporate more inherent safety features. Both of these characteristics enhance the operational competitiveness of nuclear generated power. An advantage of smaller nuclear reactors is that more of the fabrication can be performed in a factory, allowing for cost savings as well as simplifying quality assurance. Most of the reactor designs and concepts discussed in this report elaborate the extent to which factory fabrication can be incorporated. Emphasis is placed on assessing the technical feasibility of each concept or design, including identifying any innovative features and highlighting any further research which might be required for successful development.

The information contained in this chapter was collected from numerous sources. Detailed material on several designs and concepts were obtained in response to the announcement of this study in the *Federal Register* and the *Commerce Business Daily* in February 2001. The depth of information obtained, however, varied. For instance, more detailed material on the RS-MHR was received in response to the announcement for information on SMRs and is reflected in the length of its review. Information on designs and concepts was also gathered through literature and Internet sources. For most of the other SMR designs and concepts reviewed, the amount of technical information is limited, mostly because the designs and concepts are in an early stage of development. As a result, this report contains no illustrations of any plant designs or concepts. A summary of SMR designs and concepts is provided in Table 1, on page 28.

Both designs and concepts for SMRs have been examined in this study. For the purposes of this report, “designs” are those power plants where the engineering has been completed, and the plant is ready to be, or already has been, constructed. “Concepts” are power plants where the basic layout of the facility has been established, perhaps even down to some of the smallest details, but that design tradeoffs and optimization have not been done, and the detailed engineering to build the plant has not been completed. For example, the SMR concepts that are being pursued as part of the DOE’s Nuclear Energy Research Initiative (NERI), which are discussed below, are at an earlier stage of development; they may not be available for deployment as early as the other SMRs, but they are of longer-term interest.

The International Atomic Energy Agency (IAEA) periodically surveys the design and development status of small and mid-range power reactor systems in IAEA member states[1]. Among the concepts covered in the IAEA survey, only four are in the power range of less than 50 MWe. These four international SMRs are evaluated in terms of technical merits that would impact their suitability for deployment in the United States in the near future.

In 1999, DOE started a program called NERI to promote research focused on advanced technologies for improving the cost, safety, waste management, and proliferation resistance of nuclear energy systems. One area of research being considered is the development of innovative, small, compact, and easy to deploy, power reactor designs that employ passive safety systems and long life cores. The concepts meeting the requirements for SMRs are included in this study. The common features of different small reactor designs include:

- modularity (factory fabrication of modules and shorter construction time),
- increased safety margins and reduced severe accident probability and consequences (inherent safety features that reduce or eliminate the need for engineered safety systems),
- suitability to local electrical grid requirements,
- design flexibility for applications beyond power generation (district heating and desalination), and
- lower initial capital investment.

Depending on the availability of specific information on each of the SMRs, some discussions are more extensive, while others are much briefer and are included mainly for perspective.

3.1 CAREM (Argentina)

The CAREM project by the Argentinean National Atomic Energy Commission (CNEA) and commercial supplier INVAP is based on a simplified pressurized water reactor (PWR) design with ratings of 100 megawatts-thermal (MWt) and 25 MWe. The design includes a helical steam generator above the core fed by natural circulation so the unit has no main coolant pumps or pressurizer. It has innovative, hydraulic control rod drives, a large volume primary coolant system, and a negative temperature coefficient–which is an inherent effect that simply means that the reactor power will automatically drop when there is an increase in temperature, consequently bringing power and temperature back down.

The most innovative feature of this design is that the entire primary coolant system is contained within the reactor pressure vessel, so it is termed an "integral PWR." The integral reactor vessel contains the reactor core and support structures, steam generators, and the control rod system. The primary system is self-pressurized by the steam generated inside the vessel. The operating pressure is the steam pressure corresponding to the temperature of the coolant at the core exit. A steam chamber, located near the top of the reactor vessel, is used to regulate pressure against variations in coolant temperature.

Pumps are eliminated in the primary system and core cooling is accomplished by natural convection. Using natural circulation instead of coolant pumps has a number of important benefits contributing to higher reliability and safety, better economic performance, and sabotage and proliferation resistance. The primary coolant flow rate and circulation pattern is determined by the elevation difference between the core (heat source) and the steam generator (heat sink), and the rate at which heat is removed at the steam generator. Each steam generator is a standard shell-and-tube type heat exchanger. The primary coolant flows downwards through the tube side while the secondary coolant flows upwards through the shell side where it is preheated, evaporated, and superheated. In addition to the steam generators, the secondary system consists of a turbine, condenser, aerator, feedwater system, bypass line, and control valves. The auxiliary systems include primary and secondary water supply and treatment systems, chemical and volume control systems, component cooling systems, a refueling system, and a spent fuel storage pool. The steel reactor containment is of the pressure suppression type, supported by a system to condense escaped steam and keep the pressure inside the containment within design limits.

Fuel Characteristics

CAREM uses standard PWR fuel technology with 3.4 percent enriched uranium-oxide fuel contained in fuel pins. These materials are not attractive for use directly in weapons, and provides the first level of defense for diversion.

Safety Aspects

Even though many light water reactors relying on inherent safety from the negative temperature coefficient, use soluble boron in the coolant water for reactivity control, the CAREM core design does not during normal operations. Instead it uses burnable poison, a neutron absorber, inside the fuel, that is consumed as the reactor operates, to offset the loss in reactivity from consuming the fissionable uranium. The reactor can be fully shut down using 24 neutron absorbing control rods. There is also a backup boron injection system, in case additional neutron absorption is required, if the control rods fail to insert. The control rod drive system is of the hydraulic type and is almost entirely contained within the pressure vessel. Once the reactor is shut down, any residual heat generated in the core is removed by natural circulation. In situations where steam generators cannot function as a normal heat sink, passive heat removal is achieved through two redundant gravity-driven systems that inject water in the steam generators.

The possibility and consequences of a Loss of Coolant Accident (LOCA) are greatly reduced since the entire primary system is enclosed within the reactor pressure vessel. Vessel penetrations below the core, such as piping connections or instrumentation locations are avoided, assuring that a large inventory of water is always available for passive cooling in the event of a break. The design also includes an emergency core cooling system as additional protection against a LOCA. The system consists of two redundant water-filled tanks pressurized by nitrogen gas, providing sufficient volume and flow rate to ensure that the core will be kept underwater for an extended period.

The elimination of pumps in the primary system simplifies the design substantially, adding to safety. Specifically, any Loss of Flow Accident (LOFA) scenario associated with coolant pump failure is eliminated as a possible accident initiator. Elimination of pumps also facilitates factory fabrication of some components as modules, and simplifies the operation, inspection, maintenance, and quality assurance requirements. The elimination of large primary system coolant pipes, and hence of the need to contain the rapid pressure increase in the containment associated with a primary coolant pipe break, allows an easing of the containment specification.

Diversion Resistance
One of the main disadvantages of this design is its annual refueling requirements. Since only 50 percent of the core is withdrawn during refueling, it is performed on site. This may constitute a diversion or proliferation concern since refueling requires frequent transportation and handling of nuclear fuel, and on-site spent fuel storage.

Overall Assessment and Potential Issues
In summary, CAREM is a mature design. Its high pressure natural circulation capability, control rod mechanism, and neutronic data have all been validated at test facilities. CNEA currently has legislative authorization from the Argentinean Government to seek financing and provincial approval for a site to build a prototype. The CAREM design can be ready for deployment within this decade. Since the reactor fuel is uranium oxide with low enrichment, the fuel manufacturing, spent fuel processing, and waste disposal technologies are the same as for existing large commercial light water reactors. In addition to the annual refueling requirement, another disadvantage is that the CAREM power plant is not highly modularized, requiring a substantial amount of on-site construction.

3.2 ENHS (United States)

The Encapsulated Nuclear Heat Source (ENHS) is a concept being developed under the NERI program by a consortium lead by University of California-Berkeley. It is a liquid metal-cooled reactor (LMR) that can use either lead (Pb) or a lead-bismuth (Pb-Bi) alloy as the reactor coolant. As opposed to sodium as the traditional liquid-metal coolant, the lead-based coolants are chemically inert with air and water, have higher boiling temperatures, and better heat transfer characteristics for natural circulation. The ENHS has a very long core life, and it uses natural circulation to cool the reactor core and to produce steam to drive the turbine. The ENHS concept relies on autonomous control, that is, after the reactor is brought to full power, variation in power output follow the electricity generating needs automatically (load following) by using temperature feedback from the varying steam pressure and feedwater flow.

The ENHS concept is based on the idea of encapsulating the reactor core inside its own vessel as a module, with no external piping connections. The core is located in a central vertical cylinder inside the vessel. The annular region between this cylinder and the outer wall of the reactor module is constructed as a counterflow heat exchanger. This ENHS module is inserted into a large pool of secondary molten metal. The heat generated in the core is carried upward by the primary molten metal coolant to the top of the vertical cylinder, where openings connect to the primary side of the annular heat exchanger region. The primary coolant flows downward and back through another set of openings under the reactor core. The molten metal in the pool enters the secondary side of the annular heat exchanger through openings in the reactor vessel at the bottom, and exits through another set of openings at the top. In this manner, the heat generated in the core is transferred to the secondary pool passively through the counterflow heat exchanger in the reactor vessel without using any piping connections.

The steam generators are separate modules which are also inserted into the secondary pool, adjacent to the reactor vessel module. The molten metal in the pool enters the pool side of the steam generator through openings near the top of the steam generator, and exits near the bottom of the steam generator after having transferred heat to the water in the steam generator. Water also circulates through the steam generator using natural circulation, so that no pumps are used in the entire reactor system. The concept can automatically load follow over a wide power range. The use of small steam generators makes it easier to design the power plant to use supercritical (very high temperature) steam and, thus, thermal-to-electrical energy conversion efficiencies exceed 42 percent. The secondary coolant pool design also offers the flexibility to connect to desalination plants or district heating systems.

Fuel Characteristics

The ENHS fuel is a metallic alloy of uranium and zirconium (U-Zr) or optionally uranium, plutonium, and zirconium (U-Pu-Zr), and exhibits good stability under irradiation. The fuel is contained in cylindrical fuel pins with a large fission gas plenum above to accommodate high burnup of the fuel and the resulting expansion from gaseous fission products. The reactor can operate at full power for 15 years using either U-Pu-Zr metallic fuel having about 11 percent plutonium, or U-Zr metallic fuel using uranium enriched to 13 percent U^{235}. The core consists of fuel assemblies having 217 rods in a hexagonal array with the central location reserved for a large safety element, which can assure complete reactor shutdown. The core is surrounded by six groups of tungsten segment reflectors.

Safety Aspects

The ENHS concept is characterized by a large thermal inertia because of the large inventory of the primary and secondary liquid-metal coolant, making the concept inherently safe. In all accident sequences, heat can be transferred to the vessel boundary by conduction and natural convection while the fuel and cladding temperatures remain significantly below safety limits.

Diversion Resistance

The ENHS can operate at full power for 15 years. The ENHS module is manufactured and fueled in the factory, and shipped to the site as a sealed unit with solidified Pb (or Pb-Bi) filling the vessel up to the upper level of the fuel rods. With no mechanical connections between the reactor module and the secondary system, the module is easy to install and replace, similar to using a battery. After installation, hot coolant is pumped into the vessel to melt the solid lower part. At the end of its life, the ENHS module could be removed from the reactor pool and stored on site until the decay heat drops to a level that lets the coolant solidify. The module with the solidified coolant would then serve as a shipping cask. Its compact, sealed design combined with very infrequent refueling provides high proliferation resistance. These design characteristics are intended specifically for remote siting; however, the total weight of an ENHS module when fueled and when loaded with Pb-Bi to the upper core level is estimated to be 300 tons, which could pose a shipping challenge, especially to remote areas.

Overall Assessment and Potential Issues

The ENHS concept offers a safe system that is characterized by low waste, high proliferation resistance, high uranium utilization, and simplicity of operation. Since the ENHS is only at the conceptual stage, it is not likely to be ready for deployment in this decade. While the ENHS relies on LMR experience, the use of lead or lead-bismuth alloys instead of sodium for the reactor coolant poses significant technical challenges, especially in the compatibility of the molten lead or lead-bismuth coolant with structural materials, such as steel, at high temperatures.

3.3 IRIS-50 (United States)

The International Reactor Innovative and Secure (IRIS) concept is being developed under the NERI program by an international consortium led by Westinghouse Electric Company. IRIS is a PWR designed to address the requirements of proliferation resistance, enhanced safety, improved economics, and waste reduction. IRIS-50 is a concept variant using a low power rating (50 MWe) and natural circulation.

The IRIS-50 concept is based on proven light water technology with innovative modifications. One of the notable IRIS-50 characteristics is the integral reactor vessel. The reactor vessel and other components are also surrounded by a steel containment, spherical in shape and estimated to be about 16 meters (m) to 18 m in diameter.

Fuel Characteristics

The reactor core has 21 fuel assemblies and a diameter of 1.5 m. The active core height is 1.8 m. The reference IRIS-50 design features a five-year refueling interval using uranium oxide fuel with an enrichment of 5 percent U^{235}. Higher enrichment would allow a refueling interval nearly double (nine years), but the higher fuel burnup would require additional testing and analysis for licensing approval. Both the fuel materials and the fuel assembly design are similar to the current commercial PWR technology. For this reference fuel design option, there is an extensive amount of relevant fuel fabrication experience. The reduced power density provides an additional safety margin compared with large commercial PWRs.

Safety Aspects

Burnable poison is included in the design to maximize the negative temperature coefficient, which is an inherent safety feature. Since the control rod system is partially contained within the reactor vessel, control rod ejection is eliminated as an accident initiator that would lead to an uncontrolled power increase. This is a safety improvement compared with existing PWRs. The IRIS-50 concept features external control rod drive mechanisms, but remotely controlled electromagnetic drives or pressure actuated hydraulic drives, similar to the CAREM design are also being considered to achieve containment of the control rod system fully inside the reactor pressure vessel.

The low power density core of the IRIS-50 permits the elimination of the primary system coolant pumps for forced circulation, eliminating the LOFA scenario associated with coolant pump failure as a possible accident initiator. On the other hand, the primary system requires a taller vessel than would otherwise be needed if coolant pumps were used. The IRIS-50 reactor vessel is currently estimated to be 3.5 m, outside diameter, and 14 m to 16 m, that may affect its transportability.

The probabilities of steam line breaks and steam generator rupture accidents are minimized by having the steam generator operate at the same design pressure and temperature as the reactor vessel. A small leak can be controlled with isolation valves eliminating the need for steam line safety valves. The probability of a small-break LOCA is also reduced because of fewer reactor vessel penetrations, but in the event of a small LOCA, the high pressure containment design and the efficient in-vessel heat removal system allow the core to remain covered for an extended period of time without the need for additional water makeup. The efficient heat removal is achieved by condensing the steam within the vessel and returning water directly to the core to limit the vessel pressure. The large water inventory inside the reactor vessel, combined with the elimination of piping loops that could break, further simplifies the plant as compared with existing PWRs. Following shutdown of the core, any residual heat is transferred out of containment by natural circulation to heat exchangers submerged in a water tank.

Diversion Resistance
The IRIS design is based on the well-proven PWR technology that has been adopted worldwide for commercial power reactors over many decades. The IRIS-50 concept is sized to have the major components transportable. With the fuel being essentially the same as commercial PWR fuel, the IRIS-50 fuel has the same diversion and proliferation resistance. The integral design of the reactor vessel improves the resistance against sabotage compared with conventional PWR designs. However, on-site spent fuel storage is needed at least for a short time. As part of another NERI project, barge mounted construction is also under consideration for transportation and operation. In this case, the core would remain sealed and fully inaccessible at the remote location and returned to a center for refueling and maintenance.

Overall Assessment and Potential Issues
In developing the overall IRIS-50 concept, international organizations have joined with Westinghouse to assist in the successful completion of the design. With commercial interest from Westinghouse and technical support from the consortium, IRIS-50 has the potential for deployment in this decade.

3.4 KLT-40 (Russia)

With its design based entirely on the nuclear steam supply system used in Russian icebreakers, KLT-40 is a proven, commercially-available, small PWR system. It is a portable, floating, nuclear power plant intended mainly for electric power generation, but it also possesses the capability for desalination or heat production. Although the reactor core is cooled by forced circulation of pressurized water during normal operation, the design relies mainly on natural convection in the primary and secondary coolant loops in all emergency modes. The plant is mounted on a barge, complete with the nuclear reactor, steam turbines, and other support facilities. It is designed to be transported to a remote location and connected to the energy distribution system in a manner similar to the Mobile High Power Nuclear Power Plant (MH-1A) operated by the U.S. Army in the 1970s, as described in Chapter 2. The designer and supplier of the KLT-40 is the Russian Special Design Bureau for Mechanical Engineering (OKBM).

Fuel Characteristics

The reactor core contains 241 fuel assemblies. Fuel is a uranium-aluminum metal alloy clad with a zirconium alloy. An inventory of 200 kg U^{235} gives a fairly high core power density of 155 kW per liter on average, and the fuel may be high enriched uranium (U^{235} content at or above 20 percent). The fuel assembly structure and manufacturing technology are proven and their reliability has been verified by the long-term operation of similar cores. There are four coolant pumps in the primary circuit of the KLT-40, feeding four steam generators. Each of the steam generators has the capacity to produce 65 tons of superheated steam at 290^{o} C and 450 pounds per square inch (psi). The secondary system of the plant consists of two turbogenerators with condensate pumps, main and standby feed pumps, and two condensers. In the steam condensers, up to 35 MWt energy can be transferred to a desalination plant via an intermediate circuit. The design of the KLT-40 includes a steel containment vessel capable of withstanding over-pressure conditions. There is also a passive-pressure suppression system for condensing steam which escapes from the KLT-40 system into the containment building.

Safety Aspects

The inherent safety characteristics of the KLT-40 include a large negative temperature coefficient for the reactor core, where increasing core temperature lowers core power. This is achieved in the KLT-40 design without the use of soluble boron in the coolant water. Instead, a large quantity of burnable poison is used in the fuel and more control rods are incorporated in the design to ensure a cold shutdown. A large negative temperature coefficient can be a concern for any over cooling events, such as a steam-line break or inadvertent operation of emergency cooling systems. Consequences of such events are reduced by providing a large reactor coolant inventory, thus limiting the maximum rate at which the core power can change.

The reactor safety system is designed as an active system that relies on gravitational acceleration of spring-loaded control rods in response to an initiating event. After the reactor is shutdown, the residual heat in the core can be removed from the primary coolant through the secondary system. In case of loss of off-site power, the residual heat can be removed by natural convection in all coolant loops. The large volume of cooling water in the KLT-40 gives the operator a longer time to analyze emergencies and organize accident management compared with a conventional PWR.

An emergency core coolant system provides water in case of a LOCA. A feedwater system with three pumps is also provided to compensate for small leaks and, if necessary, to inject a liquid absorber into the core. Since the reactor core is maintained under water, however, the design has the capacity to remove the residual heat passively following a primary circuit break. The integrity of steel containment is maintained by two redundant self-actuated emergency pressure suppression systems, which keep the pressure inside the containment within design limits.

Diversion Resistance

Two of the most unique features of KLT-40 are factory fabrication and transportability over water to remote locations. Although the KLT-40 requires refueling every two to three years, which may be less than desired, the transportability of the entire plant to maintenance centers provides enhanced proliferation resistance. DOE and Navy experience with aluminum-based fuels can be used in addressing issues related to fuel fabrication and handling, spent fuel processing, and waste disposal.

Overall Assessment and Potential Issues

The KLT-40 can be best characterized as a small loop-type PWR design; that is, the major components of the primary coolant system are connected by high pressure piping to form a coolant loop, through which the primary cooling water is circulated. For the most part, KLT-40 relies on vast Russian experience with similar conventional PWR technology. Compared with a large commercial PWR, its lower power output leads to minor simplification of the design and, to some degree, allows for reliance on natural processes for passive safety. However,
KLT-40 is still a PWR requiring all of the typical active engineering safety systems. The design is technically mature, having been used for many years in marine propulsion applications. Its similarity with conventional PWR power systems could be an advantage in terms of licensing readiness. As a proven concept for power plants for Russian icebreakers, KLT-40 is ready to be considered for deployment.

3.5 MRX (Japan)

The MRX design is a marine power reactor originally designed for an icebreaker and scientific observation ship. Like CAREM, it is an integral PWR with the steam generator and pressurizer installed inside the pressure vessel, although there are other major components of the primary coolant system which are outside of the reactor vessel. A relatively large water inventory increases the thermal capacity of the primary system and reduces radiation damage to the vessel. The designer is the Japan Atomic Energy Research Institute (JAERI).

Fuel Characteristics

The design of fuel elements is based on well-developed PWR fuel technology. It uses 4.3 percent enriched uranium oxide fuel contained in fuel pins. The reactor core consists of 19 fuel assemblies, 13 of which contain control rods. Six of the control rod clusters are used for reactivity control and the other seven for reactor shutdown. Since the reactor core has a low power density, the MRX responds slowly to temperature variations. The design of the reactor's core allows a cold shutdown to be assured without using a boron solution in the primary coolant water, even with one control rod cluster withdrawn. The MRX design adopts a partially-passive decay heat removal system, where the residual heat is removed from the primary coolant by means of the steam generator; however, valve movements are needed to reconfigure the steam generator into a natural circulation heat exchanger.

Safety Aspects

The pressurizer and the once-through helical-coil type steam generator with two redundant sets of feedwater and steam lines are incorporated inside the reactor pressure vessel. However, the main coolant pumps, volume control system, and a residual heat removal system are all outside the pressure vessel. Since the piping for these primary system components is small, the effect of a pipe break is reduced compared with a conventional PWR. The primary system coolant pipes penetrate the vessel near the top; thus, the core remains flooded in the event of a pipe break, giving the design some inherent safety protection against LOCA. The containment is also filled with water to provide additional inherent safety, eliminating the need for an emergency core cooling system. A water-tight containment surrounding the pressure vessel is included in the design to limit the heat loss from the primary system during normal operations.

The remainder of the MRX plant includes the main steam system, low-pressure steam generator, condenser, feedwater system, and the instrumentation and control system. The main steam system consists of the main generator turbine that provides 30 MWe output, and two service generator turbines to provide power for auxiliary machines and facilities. The low-pressure steam generator and turbine are mounted on the condenser.

Emergency core cooling is accomplished by a closed system that transfers decay heat from the core to the containment water. The system relies on natural convection between the steam generator (as the heat source) and a containment water cooler (as the heat sink). Also, provided in the design are a containment water-cooling system and a residual heat removal system for long-term passive cooling.

Diversion Resistance

Similar to the Russian KLT-40, the advantage of the MRX design comes from its assembly-line fabrication of the entire plant, and its transportability as a self-propelled ship. Again, refueling every 3.5 years may be more frequent than desired, however, the capability of performing refueling and maintenance activities at a central facility improves diversion- and proliferation-resistance. Since the design of fuel elements is based on well-developed PWR fuel technology, the design lends itself to existing commercial light water reactor infrastructure for fuel fabrication and handling, spent fuel processing, and waste disposal.

Overall Assessment and Potential Issues

With most of the research and development work and supporting tests completed, the MRX design is in the detailed design phase, with current activities focused on evaluating the overall safety, reliability, and economy. A new type of marine reactor with an integral PWR and a water-filled containment to add to safety and radiation shielding, the MRX may be considerably lighter in weight and more economical than previously constructed Japanese ship reactors of comparable size. The design has the potential for deployment in this decade.

3.6 Modular Simplified Boiling Water Reactor (United States)

The Simplified Boiling Water Reactor (SBWR) by General Electric (GE) is a concept that incorporates advances in existing, proven boiling water reactor (BWR) technology at the 600 MWe power level. As part of the licensing audit of the GE application to NRC for design certification, Purdue University has been conducting a detailed study of the SBWR safety systems through integral tests, and developing a data bank for various transient scenarios. The "Modular SBWR" is Purdue's variation resulting from their systematic study of the SBWR. The Modular SBWR is being developed under the NERI program both at 200 MWe and 50 MWe levels.

In general, BWRs are reactor designs that produce saturated steam directly in the reactor core, and use that steam directly to drive turbines for power generation. This eliminates the need for steam generators. In addition, the Modular SBWR design relies on natural circulation to cool the reactor core and to produce the steam needed to drive the turbine and generator. As a result, the coolant pumps are also eliminated from the Modular SBWR system.

Fuel Characteristics

The proposed 50 MWe Modular SBWR is a small, compact reactor concept with modifications on the fuel cycle and fuel type for extended core life and proliferation resistance. The reactor vessel is 8.5 m in height and 3.5 m in diameter. The active core height is 1.9 m, and has an inherent negative temperature coefficient and a negative void coefficient; that is, if more of the coolant in the core is in the steam or vapor phase, the reactor power will decrease, since the neutrons are not slowed down as effectively as in the liquid water phase. Using uranium oxide fuel with an initial enrichment of 5 percent U^{235}, similar to commercial BWR fuel, it is estimated that the Modular SBWR will provide full power for ten years of continuous operation before requiring refueling.

Safety Aspects

The containment of the Modular SBWR is a cylindrical steel tank with overall dimensions of 14.6 m in height and 12 m in diameter. In addition to the reactor vessel, the containment includes compartments for various safety system components. The containment is isolated and placed in the steel-reinforced concrete cavity that provides an additional barrier against the leakage of contaminated coolant.

The passive reactor safety systems consist of the gravity-driven cooling system, suppression pool, containment cooling system, isolation condensers, and the automatic depressurization system. The depressurization of the reactor vessel allows gravity-driven injection from the emergency core cooling system to protect against a pipe-break accident. Multiple natural circulation condensers have been adopted as a passive means for long-term cooling of the containment. Most of these components and safety systems are inherited from the GE's SBWR concept. Instead of active engineered safety systems, the emergency core cooling systems are based on passive gravity-induced flow. Containment cooling for decay heat removal is also performed by a passive system; therefore, a large emergency power supply is not needed. These design simplifications increase the reliability of the system while reducing its cost.

Diversion Resistance

The reduced core power density enables single fuel batch loading with 10-year core fuel life. Such infrequent refueling adds to proliferation resistance. An alternative fuel using a mixture of thorium and uranium oxide is also being investigated, and offers some advantages for proliferation resistance in terms of the relative isotopic concentrations of uranium and plutonium. The use of thorium-based fuels, however, has been limited and higher burnup experience is needed. On-site storage of the spent fuel is likely to be required, at least for the short-term.

Overall Assessment and Potential Issues

In summary, the Modular SBWR concept is simpler than conventional BWRs, using natural circulation in the primary vessel and with no need for steam generators. The small size of the major reactor components allows these components to be factory constructed and transported to remote sites. However, substantial on-site assembly may still be needed. The NRC's familiarity with the GE design of the SBWR could be an advantage for achieving the Modular SBWR design certification because the proposed 50 MWe design has many common safety system features with the original GE design. The design may be ready for deployment in this decade.

3.7 RS-MHR (United States)

General Atomics (GA) has proposed a concept named the Remote-Site Modular Helium Reactor (RS-MHR) that is a small nuclear power plant for use in remote areas. The RS-MHR is a compressed-helium gas-cooled reactor. Gas-cooled reactors have been in development at GA for many years. Commercial examples related to this technology in the United States were Peach Bottom 1, a 40 MWe plant which operated from 1967 to 1974, and Fort St. Vrain, a 330 MWe plant which generated power from 1976 to 1989. However, in the case of Fort St. Vrain, a series of equipment problems greatly reduced plant reliability, which resulted in the early retirement of the plant. GA has incorporated lessons learned from the Fort St. Vrain experience in developing recent gas-cooled concepts, including the RS-MHR, such as eliminating the need for a steam generator and simplifying the primary coolant system. The nuclear reactor is contained in one vessel, while all of the power production and heat transfer equipment is in a second vessel, connected by a single coaxial pipe that carries the helium coolant between the two vessels.

The entire power plant and the support systems are housed in a building about four stories high at its highest, measuring about 18 m by 24 m. The building is constructed entirely above ground, eliminating the need for excavation. The reactor portion of the building is reinforced concrete–about 1 m thick for shielding purposes–and is about three stories high, measuring about 10 m by 12 m. The power rating for the reactor ranges from 10 MWe to 25 MWe.

The RS-MHR uses compressed helium gas for the reactor coolant. That helium gas is chemically inert is a significant advantage for material compatibility and corrosion resistance. The helium gas also has no effect on the nuclear reaction taking place in the reactor. The helium gas removes heat from the reactor core and drives a commercially-available industrial turbo-compressor directly. The helium turbo-compressor both generates electricity and compresses the helium before it is sent back to the reactor core. The use of helium gas allows the reactor to be operated at much higher temperatures compared with a water-cooled system, which leads to improvements in electricity generating efficiency, defined as the portion of electricity generated from the total amount of energy generated by the reactor. The RS-MHR plans for an inlet temperature of 500EC and an outlet temperature of 850EC, resulting in an overall efficiency of almost 50 percent. For comparison, water-cooled nuclear reactors, and many fossil-fired steam-driven power plants, have a generating efficiency in the range of 30 to 35 percent.

Fuel Characteristics

The RS-MHR uses uranium oxide fuel, similar to that used in most existing commercial nuclear power plants, but the fuel is contained in very small spherical particles approximately 1 mm in diameter rather than in the long fuel rods typically used in large power reactors. As shown in Figure 1, the uranium oxide fuel is at the center of each spherical particle and is surrounded by a number of layers. The thin carbon (pyrocarbon) layers provide the structural integrity for the fuel particle. The silicon carbide (SiC) layer is an extremely important diffusion barrier, intended to provide the containment function for the radioactive fission products.

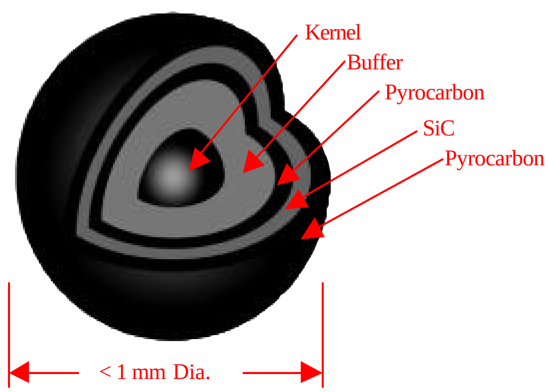

Figure 1. Pyrocarbon Coated Particle Fuel

Groups of these fuel particles, along with sufficient graphite, are formed into cylindrical fuel elements, called compacts. The fuel compacts are about 1.25 centimeters (cm) in diameter and almost 5 cm long, and are inserted into the large graphite blocks which constitute the reactor core. The graphite blocks also have holes through which the helium coolant flows. The reactor core consists of layers of these graphite blocks, and is assembled at the site once the plant is constructed. For a 10 MWe reactor core, three layers of 19 blocks each would be used, for a total of 57 graphite blocks.

The characteristics of the reactor fuel and the design of the reactor core result in a number of attractive features. First, the use of a large quantity of graphite allows for a more efficient use of the uranium fuel, contributing to infrequent refueling. Second, the large quantity of graphite and the distributed uranium fuel results in a lower power density. Although this is neither an advantage nor a disadvantage in normal operation, the low power density is a great advantage in the unlikely event of an accident, since the reactor temperature rises much more slowly and can dissipate easier than with more compact reactor core designs. Last, the design of the reactor core and fuel is identical to that which was used in Fort St. Vrain, except that the fuel enrichment for RS-MHR is much lower. Operational experience demonstrates that this is a reliable and proven fuel concept, with no unresolved technical issues.

Safety Aspects

The RS-MHR reactor core has a negative temperature coefficient. The ability to withstand very high temperatures, combined with the negative temperature coefficient, provides an inherent protection of the RS-MHR in many accident situations. For the RS-MHR, sufficient heat removal capability is planned in order to keep the peak temperatures below acceptable limits by using natural circulation in water-cooled wall panels in the reactor vessel room. Based on current estimates provided by GA, the highest temperature at any location in the core that would likely be achieved in an accident situation would be between 1500EC to 1600EC. It will be necessary to ensure that this upper temperature limit is below, by a sufficient margin, the temperature where the containment afforded by the pyrocarbon barrier could begin to break down.

The RS-MHR concept also has design features that prevent adverse conditions from developing that could be detrimental to the integrity of the reactor core. An example is the choice of location for the coaxial cooling pipe connecting the reactor vessel with the power module. One of the vulnerabilities for gas-cooled reactors constructed from graphite is the possible entry of air or water into the primary coolant system, since graphite reacts with oxygen and degrades rapidly at high temperatures. In order for air to enter the primary coolant circuit, there would have to be a failure of part of the primary coolant system boundary, such as a pipe break or vessel penetration. The only time such air entry causes a significant problem for a graphite reactor is if sufficient natural circulation flow of air through the reactor core can occur. The RS-MHR minimizes the probability of developing this condition by locating the coaxial coolant pipe below the level of the reactor core. Establishing air flow by natural circulation through the core following breakage of this pipe would be difficult, increasing the level of safety in the RS-MHR.

Safety is further improved for the RS-MHR by the elimination of the steam generator. Water entry into the primary system from the steam generator is one of the major reasons that the Fort St. Vrain reactor had operational and reliability problems. By eliminating the steam generator, the RS-MHR concept removes the major source of water that could potentially enter the primary system during an accident situation.

Diversion Resistance

The RS-MHR uses low enriched U^{235} at 19.9 percent. The refueling interval is long, estimated at six to eight years of operation. Such infrequent refueling minimizes the opportunities for access to the nuclear materials in the reactor. It also ensures that the reactor fuel is taken to a high "burnup," that is, most of the fissionable materials are already used. This is especially important for any plutonium that may be formed during reactor operation. Next, the fuel is contained in a graphite and silicon carbide. While the graphite would not appear to offer much of a barrier to diversion, the silicon carbide layer provides protection in that it is somewhat difficult to remove or penetrate. Also, the fact that the fuel is widely distributed throughout the core makes it necessary to divert a large amount of core materials to obtain a small amount of nuclear materials. The relatively small number of fuel blocks in the core should make theft or diversion of the blocks more readily apparent.

The reactor is refueled at the site, with the spent fuel being stored in a small room adjacent to the room containing the reactor vessel. GA estimates that the spent fuel will need to be water cooled for about six months, after it has been removed, then it can be dry stored, but with an active cooling system.

Overall Assessment and Potential Issues

The RS-MHR makes use of the inherent capabilities of the fuel and core materials to provide acceptable response to a variety of potential accidents. The designers have also taken advantage of the lessons learned from previous efforts, and reduced or removed previous vulnerabilities. There are also characteristics which make diversion and proliferation unlikely, and sabotage difficult. The long refueling interval reduces the handling of nuclear materials. While the fuel fabrication issue does not appear to be a significant barrier to development or deployment of the RS-MHR, the construction of a commercial fuel fabrication facility or the modification of an existing one will be necessary.

Parts of the RS-MHR concept represent a substantial deviation from practice at existing nuclear power plants. Specifically, the RS-MHR does not have a containment building, whose functions are to contain radioactive materials that might escape from the reactor system and to withstand pressurization as the result of an equipment failure or accident. GA describes the silicon carbide layer on the fuel particles as the "containment," and asserts that no further containment is necessary. It remains to be proven that the fuel will provide the containment function for all conceivable situations in the reactor plant, including hypothetical accident conditions. The design may be ready for deployment at the end of this decade.

3.8 TRIGA Power System (United States)

The General Atomics (GA) TRIGA Power System (TPS) is a PWR concept based on the TRIGA reactor design coupled with a commercially available organic power system. GA has been developing the TRIGA line of reactors since 1958, mainly as research reactors at universities, hospitals, and research institutions throughout the world. According to GA, the original concept of using a reactor based on the TRIGA experience in combination with a power generation system was initiated in the mid 1980s in connection with a U.S. military program for secure remote power supplies. The TPS is designed for a power level of 64 MWt, 16.4 MWe.

The TPS measures 40 m by 60 m. There is also excavation required below grade, to a depth of about 10 m. The TPS building is a confinement building having concrete reinforced walls, a floor, and a roof about 1 m thick. Such a confinement provides radiation shielding, but the sufficiency of using no containment would have to be validated.

Like standard PWRs, the primary coolant system of the TPS consists of the reactor core, primary circuit piping, pressurizer, coolant pump, and a heat exchanger. The reactor vessel containing the core and the primary heat exchanger where heat is transferred from the primary circuit to the secondary circuit constitutes two large factory-fabricated modules to permit a transportable system. Not being an "integral PWR," there is a substantial amount of auxiliary equipment and piping systems needed to support the TPS reactor. The secondary system is housed in a room adjacent to the reactor room.

The reactor in the TPS uses a pool design, where the reactor core is located in, and physically separated from, a larger pool of water. The TPS operates at a lower pressure than a standard large PWR. The operating temperatures are correspondingly lower, with a nominal outlet temperature of 216EC (420EF). The water that cools the reactor is used to heat an organic fluid (a commercially available inert perfluorocarbon FC-72) on the secondary side of the plant, which in turn is used to drive a turbine.

While most of the features of the primary coolant circuit are conventional and well proven, and require no additional discussion, there is one aspect which is not traditional. In an effort to improve the inherent safety of the TPS reactor, the core has been placed in a pool inside the reactor vessel. During normal operation, only a small amount of the water in the reactor vessel is actually circulated to transfer heat to the heat exchanger. The remainder of the water in the reactor vessel, about 90 percent of the reactor vessel volume, is maintained at a constant temperature of about 71EC by the auxiliary cooling system that operates continuously using natural circulation. The innovative feature of this concept is the connection between the circulating primary coolant and the pool of water in the reactor vessel, termed the venturi pressure balancing system, that does not use any valves to control the coolant flow between the pool and the circulating coolant.

Fuel Characteristics

The reactor core uses standard uranium-zirconium hydride fuel, as in the TRIGA reactors. The fuel also contains a small amount of burnable poison. The TPS fuel uses low enriched uranium, with U^{235} at 19.9 percent enrichment, and is currently manufactured for the TRIGA research reactors currently in operation. The fuel rods are much shorter and smaller in diameter compared with standard PWR fuel elements. The 25 fuel rods are placed inside a square shroud constructed from Zircaloy to form a standard TRIGA fuel assembly which has been proven through many years of operational experience.

The TPS core consists of 76 fuel assemblies surrounded by 36 beryllium reflector elements. The reactor core is loaded into the reactor vessel once all of the equipment has been assembled, including control rods, shutdown rods, fuel assemblies, and reflector assemblies. The reactor power is normally adjusted using the control rods; furthermore, the reactor also has a strong negative temperature coefficient, which rapidly reduces power with an increase in temperature. These characteristics of the reactor core give the design the capability for autonomous control of reactor power. The negative temperature coefficient is an important safety feature, allowing the TPS reactor to tolerate a wide range of upsets, including accident conditions, with less chance of damage to its fuel, thus offering more protection from release of radioactivity to the environment.

Safety Aspects

The TPS is susceptible to the accidents that are common to water-cooled reactors; however, the TPS design mitigates the effects of some of these accidents, by operating at much lower pressure and temperature. To address the depressurization accidents, the design minimizes the amount of water available to flash to steam. It does appear likely, however, that the intervention of some engineered safety system would be required to shut down the reactor. In this sense, the TPS would not rely entirely on passive safety features to mitigate the accident consequences.

To respond to some accident conditions, redundancy is built in the system. The primary circuit is connected to the pool in two locations, at the top of the outlet plenum above the core, and in an adjustable venturi nozzle in the inlet pipes. During an accident, the pool water enters the core through openings in the inlet pipes and flows through the core. The heated pool water then exits through openings at the top of the outlet plenum. Since this flow path completely bypasses the remainder of the primary coolant system, decay heat from the core can be removed by the auxiliary cooling system. During normal operation, however, the static pressure equilibrium between the primary system piping and the pool at both the top of the outlet plenum prevents mixing of the coolant in the pool and the primary coolant loop. In principle, this approach provides a passive backup for decay heat removal, along with a long response time, by using the pool water. The proposed venturi pressure balancing system has not yet been proven to achieve this goal.

Diversion Resistance

The LEU standard TRIGA fuel is not attractive for use directly in weapons. Also, the reactor vessel is isolated from any routine operations. For the reference core design, one-half of the core is refueled on-site every 18 months, which is considered a short refueling interval. The spent fuel is stored inside the reactor vessel until it is cold enough to be removed and shipped from the reactor site, making access to these materials difficult.

Overall Assessment and Potential Issues

Overall, while it is accurate to describe the TPS as a combination of existing technologies, there is sufficient uncertainty about the innovative features of the reactor system that may impact the vendors ability to design and deploy the system this decade. The long operational history of the TRIGA reactors demonstrates that this is a proven fuel concept. However, the low pressure drops through the reactor core at normal operating conditions pose a challenge for assuring the desired flow distribution through the core. Also, the viability of the venturi pressure balance system requires a testing program to confirm and optimize the design of the nuclear heat source system.

3.9 4S (Japan)

The 4S is a LMR, using sodium as the coolant. The 4S design is based on the principles of simplified operation and maintenance, improved safety and economics, and proliferation resistance. The 4S design combines infrequent refueling, about every ten years, with a short construction period based on factory fabrication. The designer is Central Research Institute of Electric Power Industry (CRIEPI), Japan.

The primary coolant system includes an electromagnetic pump to pressurize the liquid sodium coolant and an intermediate heat exchanger, both placed inside the reactor vessel and above the core. The secondary system consists of another electromagnetic pump, a helical-tube type steam generator, and a sodium-water reaction product release system. The balance-of-plant systems include the nuclear steam supply system, a turbine generator, and heating and ventilation systems. There is a containment vessel that envelops the reactor vessel and the top dome.

Fuel Characteristics

The reactor fuel uses a metallic alloy (either U-Zr or U-Pu-Zr) which has been developed in the United States, and later in Japan. In the 4S reactor core, the steady-state power level is maintained throughout the core life primarily by slow vertical movement of a graphite reflector surrounding the core, rather than by using neutron-absorbing control rods. Thus, the reactor power is controlled by allowing more neutrons to leak out of the core (i.e., to not be reflected back into the core), rather than by absorbing more neutrons in the core using control rods. Even though the method of using a movable reflector is unconventional, the ability to control the reactor power is the same as using control rods.

Safety Aspects

The 4S design is a small reactor designed to have totally passive safety systems that do not require power and may not require valve movements to initiate them. Unlike helium-cooled reactors, where the helium gas has no effect on the reactor power, and water-cooled designs, where the presence of water is required for the reactor to function, a sodium-cooled reactor can be more reactive without the coolant in the core unless the “sodium void coefficient” is negative. A negative sodium void reactivity coefficient is achieved in the 4S design by keeping the core diameter small, thus enhancing the radial neutron leakage. The fuel temperature coefficient is also negative, so that reactor power inherently decreases with increasing temperature. Load following is achieved in an innovative way by controlling the water flow to the steam generator, thus manipulating the core inlet temperature. That is, as the generator output matches the load, changes in the coolant temperature introduce a positive or negative reactivity effect in the core, causing the reactor power to follow. This feature greatly simplifies operation of the 4S power plant.

The use of a movable reflector to control neutron leakage and thus the reactor power is perhaps the most unique feature of this concept. Liquid sodium is a coolant with excellent heat capacity, very high thermal conductivity, low-operating pressure, and superb natural convection capability. Decay heat is removed from the core by natural circulation of the primary coolant, and discharged by a coil system placed above the intermediate heat exchanger. If the main pump fails, however, a passive cooling is also provisioned using natural circulation of air from outside the guard vessel.

Diversion Resistance

While the presence of plutonium in the fuel may be considered a proliferation and diversion risk, the fuel is always in a highly-irradiated form, providing a level of self-protection. The fuel is also handled remotely, so that there is never any direct physical contact between the fuel and plant personnel. This physical separation enhances diversion resistance.

Overall Assessment and Potential Issues

In summary, although the 4S concept is at the basic design stage, sodium cooled reactor technology is well developed. The 4S design is an evolutionary use of proven concepts, and as such, there would not appear to be any technical barriers to this design and its deployment. With 10 years continuous operation without refueling, greatly simplified operation with autonomous control, and a short construction period based on factory fabrication, it is especially suited for remote sites. The design is in the early stages of development and may not be ready for deployment in this decade.

3.10 Other Small Modular Reactor Concepts

Two other small reactor concepts being developed under DOE programs are the Multi-Module Reactor (MMR) design from Sandia National Laboratories and the SSR design from Oak Ridge National Laboratory. Although both designs are essentially at a pre-conceptual design stage, therefore less information is available on them, they are included in the report for completeness.

- **MMR**: The MMR concept consists of an array of self-contained, factory-built, transportable gas-cooled modules in a pool configuration. The modules consist of a reactor core and an integral direct cycle turbine-compressor-generator system, all contained in a single tubular pressure vessel. The individual modules are subcritical; therefore, an array of modules in a predetermined grid structure is needed to achieve criticality. Each module is expected to have one to five MWt capacities.

- **Solid-State Reactor (SSR)**: The SSR is novel concept to achieve demand-driven heat generation without the need for moving parts or working fluids. It is a self-regulating 3 MWt nuclear heat source for small power units based on advanced graphite-foam material. This foam material, currently being studied at ORNL, has enhanced heat transfer characteristics and good high-temperature mechanical properties. The SSR concept is being developed under the NERI program.

Table 1. Summary of Small Modular Reactor Designs and Concepts

		CAREM	**ENHS**	**IRIS-50**	**KLT-40**	**MRX**	**MSBWR**	**RS-MHR**	**TPS**	**4S**
Designer		CNEA	UCB	W	OKBM	JAERI	GE/ Purdue U.	GA	GA	CRIEPI
Type		Integral PWR	LMR	Integral PWR	PWR	Integral PWR	BWR	HTGR	PWR	LMR
Rating		25 MWe	50 MWe	50 MWe	35 MWe	30 MWe	50 MWe	10 MWe	16.4 MWe	50 MWe
Primary System Pressure		12.3 MPa	N/A	-	13 MPa	12 MPa	-	-	3 MPa	N/A
Reactor Vessel	Height	11 m	19.6 m	14-16 m	3.9 m	9.4 m	8.5 m	8 m	11.6 m	23 m
	Diameter	3.1 m	3.2 m	3.5 m	2.2 m	3.7 m	3.5 m	3.4 m	2.8 m	2.5 m
Reactor Core	Height	1.4 m	1.25 m	1.8 m	0.95 m	1.4 m	1.9 m	3.6 m	1 m	4 m
	Diameter	1.3 m	2 m	1.5 m	1.2 m	1.5 m	3.1 m	3 m	1 m	0.8 m
Avg. Power Density*		55 kW/l	6 kW/m	13 kW/m	155 kW/l	42 kW/l	8.3 kW/m	4 kW/l	95 kW/l	61 kW/l
Fuel/Type		UO_2 pins	U-Zr metal	UO_2 pins	U-Al alloy	UO_2 pins	UO_2 pins	UO_2 particles	UZrH pins	U-Zr metal
Fuel Enrichment		3.4 %	13 %	4.95 %	-	4.3%	5 %	19.9%	19.9%	~15 %
Refueling Frequency (Percent Replaced)		~ 1 year (50%)	15 years (100 %)	5-9 years	2-3 years (100%)	~ 4 years (50%)	10 years	6-8 years	1.5 years (50%)	10 years (100%)
Coolant flow rate		410 kg/s	0.51 m/s	-	722 kg/s	1250 kg/s	620 kg/s	-	419 kg/s	633 kg/s
Core Inlet Temperature		284 °C	400 °C	-	278 °C	283 °C	279 °C	500 °C	182 °C	355 °C
Core Outlet Temperature		326 °C	550 °C	-	318 °C	298 °C	14.3% quality†	850 °C	216 °C	510 °C

* the amount of power generated in a given volume of the reactor core kW per liter, or power in a given length kW per meter.
† BWRs measure performance in terms of steam quality (percent by weight of vapor versus liquid) at the core outlet
"-"= Not Provided
N/A = Not Applicable

4 REGULATORY ISSUES

This chapter highlights some of the general regulatory issues related to SMR designs. Because the regulatory concerns associated with LWRs have been studied extensively by the NRC and have been the primary focus of their regulation, this chapter, by considering gas-cooled reactors, offers a starting point for a regulatory approach for SMRs consistent with current NRC risk-based techniques and safety cornerstones. Additional background information regarding the fundamentals associated with the current regulatory environment can be found in the Appendix.

4.1 Small Modular Reactor Licensing Considerations

The design of safety-related systems for a SMR that differs from current LWRs will require the development of a new licensing bases and its associated criteria using risk analysis methods, in order to take full advantage of more passive and inherent safety features. As an example, some SMR designs use compressed inert gas as a cooling medium and eliminate the high pressure and temperature steam turbine for generating electrical power, instead driving a gas turbine generator directly. Contrary to steam systems, the compressed inert gas has relatively little stored energy to release following an possible accident. Also, the decay heat removal system for non-LWR reactor designs is completely different from a LWR's.

Small gas-cooled reactors generally use different methods for protecting the environment from plant radioactivity, even after an accident, by using multiple barriers to provide "defense-in-depth." Both lower power densities and advanced fuel fabrication techniques, which can produce high-quality pyrocarbon coated particle fuel capable of withstanding temperatures higher than those encountered during accident situations, would allow the pyrocarbon fuel itself to serve as the primary barrier for the retention of radioactive fission products. As a consequence, the reactor building for such designs is intended to function as a radiation shield and not to provide the traditional containment functions of withstanding pressure increases and containing any radioactive materials.

Advanced SMRs utilize passive and inherent safety systems for accident prevention and mitigation. As an example, in many cases, no initial operator intervention is required for decay heat removal. Therefore, typical advanced reactor accident analyses can assume that few, if any, fission products will be released under accident situations and that worker, public, and environmental safety will be maintained. Possible accidents' scenarios for advanced gas-cooled reactors are: 1) loss of forced circulation of the compressed gas coolant, 2) a rapid depressurization of the primary cooling system, causing loss of the gas coolant, 3) unintentional control rod withdrawals, and 4) failure to scram, or quickly shut down the reactor in response to system malfunction.

Most of the SMRs discussed in this report use low enriched uranium (LEU). Those reactors using pyrocarbon coated fuel particles can operate at higher temperatures and burn a much higher percentage of the fissionable fuel than do conventional LWRs, which helps to reduce residual plutonium and other highly radioactive elements in the spent fuel. Reducing the amount of long-lived radioactive waste, and the resulting long fuel cycles (along with the low enrichment level) make the fuel less desirable for proliferation.

4.1.1 Accident Evaluation

Design basis accidents (DBA) play a central role in evaluating the SMR plant designs and concepts. DBAs present a combination of postulated challenges and failure events against which plants are designed in order to ensure adequate and safe plant response. During the design process, plant response is evaluated using assumptions that are intended to provide additional margins of safety.

4.1.2 Radiological Release Issues

Source Term

The radioactive source term is the amount of radiation that would be released under the most severe accident conditions and is used in evaluating the suitability of potential reactor sites and establishing emergency planning zones (EPZ) around them. The radioactive source term defined for LWRs, published in Title 10 Code of Federal Regulations (CFR) Part 100, specify the percent of the core fission product inventory to use for the analysis. However, advanced reactor designers have proposed using a mechanistic approach (i.e., the fission product release is dependent on the detailed conditions that exist in the reactor during an accident, and for those cases where fuel failure occurs, the fuel failure characteristics of the specific fuel type are taken into account) when developing possible accidents scenarios which have the potential to cause core damage. A risk-based methodology, such as PRA, is an accepted approach for performing the accident evaluation because potential hazards can be identified and accident scenarios can be evaluated.

Adequacy of Containment

For any given reactor design, defense-in-depth uses various layers of requirements to maintain safety through multiple, diverse and complementary means. In the context of the NRC's Cornerstones of Safety, reactor safety is maintained if the defense-in-depth concept is used to ensure the integrity of the physical barriers designed to prevent fission product releases. While SMR may use new methods to mitigate the frequency and consequences of reactor accidents, these methods should demonstrate, at a minimum, a level of safety equivalent to the current generation of LWRs. SMRs, therefore, should implement the defense-in-depth concept by having a design that demonstrates multiple and independent barriers to any potential fission product release.

Offsite Emergency Response Planning

Nuclear facility licensees are required by NRC regulations to develop emergency response plans. Portions of these regulations require the licensees to coordinate their plans with state and local agencies to protect public health and safety in the unlikely event of a significant release of radioactive materials to the environment. Protective Action Guides (PAG) were developed to assist authorities in deciding how much of a radiation hazard in the environment would constitute a basis for initiating emergency planning actions.

EPZs are designated as the areas for which planning is recommended to assure that prompt and effective actions can be taken to protect the public in the event of an accident. The fundamental basis for determining EPZs for a given reactor plant is related to the following exposure pathways:

> **Plume Exposure Pathway** - The whole body external exposure (gamma radiation) and the inhalation exposure caused by materials contained in the plume.
>
> **Ingestion Exposure Pathway** - The exposure from ingestion of contaminated water or foods such as milk or fresh vegetables.

As stipulated by 10 CFR 50.47, "Emergency Plans," the plume exposure pathway EPZ for nuclear power plants consists of an area about ten miles in radius, and the ingestion pathway EPZ consists of an area about 50 miles in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor are determined by 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 may also be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MWt.

Remote Siting Issues
Constructing a SMR on a remote site includes potential advantages and disadvantages. While the population in the vicinity of such a power plant would be small, it may also be difficult to address all emergency requirements, especially in difficult terrain where access may be limited. Issues regarding plant infrastructure, public health and safety in emergency situations, environmental impact, site security, and proliferation concerns must still be considered.

4.1.3 Licensing and Regulatory Evaluation for Small Modular Reactors

The traditional concerns associated with LWR designs have been studied extensively and, therefore, this section chiefly concentrates on the regulatory issues associated with a typical gas-cooled reactor, highlighting the potential regulatory changes necessary to accommodate advanced gas reactor development, considering the technologies passive and inherent safety features.

Advanced Gas Reactor Accident Mitigation Capabilities
Advanced gas reactor core size is generally designed to facilitate passive decay heat removal by conduction and thermal radiation following accident events. Other inherent safety features, such as the negative temperature coefficient of the reactor core, will decrease power to limit fuel temperature and shut down the reactor. Fuel temperatures during accidents should not exceed design failure temperatures. Therefore, the decay heat removal features, the negative temperature coefficient of the

reactor core, and the integrity of the pyrocarbon-coated particle fuel should prevent radioactive release following any of the following postulated accident scenarios: 1) permanent Loss of Forced Circulation (LOFC), 2) rapid depressurization (loss of helium), 3) continuous control rod withdrawal accident, 4) Anticipated Transient Without Scram (ATWS), and 5) failure of the primary coolant loop precooler.

Reliance on fuel integrity to limit fission product release necessitates validation testing of the particle ceramic coating to ensure that radioactive releases or fuel failures will not be caused by projected accident temperatures. In addition, if air can be drawn into the core during a rapid depressurization event, a graphite fire may result if the air flow and temperature are within the necessary ranges. Therefore, the graphite fire issue should be reviewed to ensure that graphite fires are not credible and that a traditional reactor containment building would not be required to prevent fission product release to the environment.

Source Term

Radioactive source term defines the amount of radiation that would be released to the environment after an accident. The NRC has stated [2] that the staff believes that source terms can be developed for advanced reactor designs using mechanistic analysis, provided that: 1) source terms are used in conjunction with dose guidelines consistent with those applied to LWRs, 2) the mechanistic analysis events are selected to bound credible severe accidents and design dependent uncertainties, and 3) the performance of the reactor and fuel under normal and off-normal conditions are sufficiently well understood to permit mechanistic analysis.

Adequacy of Containment

Typical advanced gas reactor designs allow for simple construction of the reactor confinement building that provides a shielding function only. A traditional containment building is not considered necessary because retention of fission product releases is accomplished by the fuel particle coating, the core design, and the reactor coolant piping.

The NRC staff believes that the following issues should be addressed to ensure the adequacy of the confinement integrity [2]: The design should ensure that adequate protection exists so that accidents with the potential to affect the fuel integrity and result in large radiation release are considered as part of the plant's design basis. In addition, fuel fabrication and performance requirements will need to be developed, consistent with the role that the fuel particles play in providing containment. The fission product retention capability of the fuel coating will need to be demonstrated over a range of operating and accident conditions consistent with the design-basis assumptions. The fuel quality assurance program will need to be verified to prove that the fuel can be manufactured to the required specifications. These performance issues should be demonstrated through development of a full-scale prototype facility. Finally, if the proposed confinement building uses independent passive heat removal systems, the adequacy of these systems should be demonstrated under worst-case conditions.

Offsite Emergency Response Planning

Typical advanced gas reactors still need to apply the NRC's defense-in-depth framework. Application of this framework indicates that changes to current offsite emergency response planning requirements are warranted because the potential consequences of severe accidents are substantially different from those for current reactors. As a result, the ceramic fuel may retain fission products sufficiently so that a power plant's EPZ can be contained completely within the plant site boundaries. In order to justify this type of change to the current offsite emergency response planning basis, the following issues need to be addressed: 1) determining the probability level, if any, below which accidents will not be considered for emergency response planning, 2) providing for increased safety in one part of the defense-in-depth framework to justify reducing requirements in another part, and 3) the acceptance of such changes by the Federal, State and local agencies responsible for offsite emergency response planning.

Remote Siting Issues

The modular constructed components of SMRs reduce the complications that would be involved with remotely siting a conventional plant. Plant components could be manufactured in a factory and transported to the site by trucks, ships/barges, air, or other methods depending on the site topography. The availability of qualified operations and maintenance staff is addressed by the simplified design, design safety features, and automated controls.

4.2 Small Modular Reactor Licensing Framework

4.2.1 Nuclear Regulatory Commission's Cornerstones of Safety Concept

As noted in the Appendix, the NRC's cornerstones of safety focus on three strategic performance areas: 1) reactor safety, 2) radiation safety for both plant workers and the public, and 3) protection of the plant against sabotage or other security threats. These strategic areas and the associated cornerstones of safety should be used for any new reactor plant design. However, the methods used to satisfy the individual cornerstones of safety do not necessarily have to match those used by current LWR designs. For example, using a pressure-retaining containment building may not be necessary if a given design's fuel coating can be demonstrated to provide an adequate barrier to the release of fission products under all DBA conditions. The following narrative discusses how SMR designs may fit within the various cornerstones of safety.

Initiating Events - Most SMR designs and concepts are simpler than existing LWR designs. This reduces the number of systems required to provide and support the heat transport and electrical generation functions of the plant. In addition, inherent safety features reduce the number and complexity of accident mitigation systems. The resulting reduction in mechanical components and associated control systems greatly reduces the potential for equipment failures that could lead to plant shutdowns, large changes in the plant's power output, or accidents.

Mitigating Systems - SMR designs typically take a different approach to mitigating accidents by using the design to reduce the potential for an accident occurring and to reduce the severity if one does occur. For example, a negative temperature coefficient is maintained for the reactor core, and passive and inherent safety systems are used to remove the human error element that can potentially affect proper plant response to accident conditions.

Barrier Integrity - Some SMR designs, rely on the integrity of the fuel to retain fission products under all postulated conditions instead of relying on a pressure-retaining containment building to contain any fission products released as the result of a reactor accident. This makes verification of fuel integrity of extreme importance because, unlike a containment building that can be periodically leak-rate tested, verification of fuel integrity after the initial fabrication would be difficult. However, if fuel performance can be guaranteed, the SMR could be much simpler and easier to maintain through the elimination of a conventional containment building.

Emergency Preparedness - An SMR would still be required to have comprehensive emergency plans to respond to a possible accident. However, the extent of the emergency plan would be based on the worst-case source term for radioactive release estimated by the accident analysis. It is possible that evacuation of the public beyond the site boundaries would not be necessary because of the estimated small-source term.

Occupational Radiation Safety - NRC regulations set a limit on radiation doses received by plant workers, and these processes would not change significantly for SMR designs.

Public Radiation Safety - Addresses the radioactive releases from a nuclear plant during normal operations. These processes would not change significantly for SMR designs.

Physical Protection - Nuclear plants are required to guard vital plant equipment. While SMR designs have been developed to be proliferation resistant, siting plants in remote locations may increase the security risk.

4.2.2 Modification to Federal Code

An application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to the health and safety of the public. These General Design Criteria (GDC) establish minimum requirements for current LWRs. The GDC is also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such units.

Some GDCs are not applicable to small modular gas-cooled reactors based primarily on the lack of water systems, such as for primary coolant, emergency core cooling injection, containment spray, containment sump, and liquid neutron poison systems. Other GDCs, however, are applicable to gas-cooled reactor technologies and should require only minor modifications to incorporate specific gas-cooled reactor and ceramic fuel and coating terminology into the individual criteria. Some of the criteria also need an in-depth review since the requirements are oriented to LWRs. The following section provides additional details regarding GDC applicability to small modular gas-cooled reactor designs:

Criteria 1 - 5, Overall Requirements - Most of these GDCs appear to be applicable to gas-cooled reactors as written. Criterion 4 needs to address the loss of coolant accidents, dynamic effects of pipe ruptures, pipe whipping, and discharged fluids, etc.

Criteria 10 - 19, Protection by Multiple Fission Product Barriers (includes electrical distribution, electrical testing, and control room) - These GDCs appear to be applicable to gas-cooled reactors as written but may require minor modifications to incorporate gas-cooled reactor terminology.

Criteria 20 - 29, Protection and Reactivity Control Systems - Most of these GDCs appear to be applicable to gas-cooled reactors as written and should require only minor modification to incorporate gas reactor terminology. However, Criterion 23's reference to pressure, steam, and water does not apply to gas reactors. The second paragraph of Criterion 26 does not apply to gas reactors because it refers only to a liquid reactor coolant, xenon burnout and soluble boron for reactivity controls. Criterion 27 needs modification as it also references liquid neutron poison addition via the emergency core cooling system. The last sentence of Criterion 28, which addresses postulated reactivity accidents, should be updated to remove references to steam line ruptures and cold water additions. References to changes in reactor coolant temperature and pressure for Criterion 28 also needs to be addressed.

Criteria 30 - 46, Fluid Systems - Many of these GDCs appear to be applicable to gas-cooled reactors, but may require modification. However, Criteria 36 to 40 (emergency core cooling and containment heat removal systems) are not appropriately written for gas-cooled reactors. Criteria 41, 42, and 43 (containment atmosphere cleanup systems) are applicable but will need to include gas-cooled reactor terminology. Criteria 44, 45, and 46 (ultimate heat sink) appear to be applicable to gas-cooled reactors, needing only minor changes to incorporate gas-cooled reactor terminology.

Criteria 50 - 57, Reactor Containment - Some of these GDCs appear to have some applicability to gas-cooled reactors. However, Criteria 50 through 54 apply only to pressure issues associated with LWR accidents. The second sentence of Criterion 50, item (1), addresses the effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and as required by §50.44 and energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning. Criteria 51 through 55 address the pressure boundary and pressure testing functions of the containment and penetrations and appear to be only applicable to LWR containment designs.

Criteria 60 - 64, Fuel and Reactivity Control - Most of these GDCs appear to be applicable to gas-cooled reactors as written and should require only moderate modifications to incorporate terminology, especially in the area of fuel storage. Criterion 61, item 5, which addresses the prevention of a significant reduction in fuel storage coolant inventory under accident conditions, may need to be updated based on air storage methods of gas-cooled reactor ceramic-coated fuels. Criterion 64, needs updating to remove reference to spaces containing components for re-circulation of LOCA fluids.

4.2.3 Risk Techniques Applied During the Design Phase

Probabilistic Risk Assessment (PRA) is being used to design the new generation of nuclear power plants. Active components, such as pumps, valves, etc. are being specified to assure a higher level of reliability. These active components are typically combined into systems in nuclear power plants that provide a safety function. Systems with a high degree of significance for safety are designed using PRA to assure high reliability using the concepts of redundancy and diversity. Similar techniques should be developed for SMR designs, although the impact of such changes may be reduced since the entire plant is typically much simpler, with fewer components.

4.2.4 Safety-Related Equipment Specification Based on Probabilistic Risk Assessment

In 1995, NRC published a policy statement on the use of PRA. The application of PRA techniques to assess accident scenarios has been accepted as part of the technical justification for current nuclear power plant license modifications. Recently, NRC has also expended considerable effort in research on, and application of, risk-informed rule making. The primary objectives of these efforts have been to develop a risk-informed regulatory framework that will enhance safety while simultaneously reducing the regulatory burden on licensees.

In SECY-99-256, the NRC staff proposed a rulemaking plan that recommended the implementation of a new rule addressing the existing 10 CFR 50 requirements involving “special treatment rules” whose scope would be affected by risk-informing. The NRC staff performed a screening of the 10 CFR 50 regulations to determine those that would be subject to any such new rule. This approach would promote regulatory focus on those systems, structures, and components (SSC) whose functions are critical to the safety of the plant and reduce the regulatory burden on other SSCs that are of low safety significance. Such an approach would benefit existing plants as well as new plants that might be proposed. However, further screening of the special treatment rules identified in SECY-99-256 appear to be warranted. Additional screening of the special treatment rules would have the goal of eliminating those requirements that would not apply to advanced reactor designs. Thus, a distinct list of special treatment rules would have to be developed for each type of advanced reactor (e.g., light water cooled, direct-cycle gas cooled). Once the list of applicable special treatment rules is developed, PRA techniques can be used efficiently to identify the SSCs of the highest safety significance. It is expected that utilization of such a risk-informed approach would streamline the regulatory approval process for advanced reactor designs. A risk-informed approach would also have the potential to improve overall plant safety because more attention is properly focused on those SSCs that are most critical to safety.

4.2.5 Using Risk-Informed Initiatives

The basis for the risk-informed initiatives is contained in Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment (PRA) in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated July 1998. The regulatory guide provides a method to use PRA to implement risk-informed programs. Programs can be developed for risk-informed inspection, risk-informed testing, and risk-informed technical specifications (i.e., setting allowed outage times according to the risk-significance). Pilot studies have been performed that indicate that these programs can reduce regulatory burden and focus resources on the most important plant components that most contribute to safety.

- The American Society of Mechanical Engineers (ASME) Committee on Risk-Informed Inspection developed techniques to select and monitor the piping segments in commercial nuclear plants according to the risk contribution. These pilot studies indicated that the number of inspections could be reduced by more than 80 percent and at the same time safety associated with piping would increase.

- The ASME Committee on Risk-Informed Testing developed techniques to select active components (pumps, valves, etc.) according to their risk significance. The technique focuses testing on the risk significant components and the failure modes that most contribute to a degradation of safety.

- A pilot study is being reviewed by the NRC that allows plant components on the nuclear quality list to be reviewed using the PRA to determine the risk significance. The proposed technique separates components by high or low-risk significance. It has been found that some components on the nuclear quality list are of low-risk significance and lower quality requirements may be appropriate. Some components that are not on the nuclear quality list are shown to be very risk important, indicating a higher level of quality requirements. This pilot study indicates that approximately half the components on the nuclear quality list have low-risk significance. Reducing the quality requirement level of these components will lead to large savings in both capital equipment costs, and operation and maintenance costs. Increasing the quality requirements for highly risk-significant components not on the nuclear quality list will increase safety.

4.3 Small Modular Reactor Licensing Recommendations

Applicants seeking to site a SMR in the United States would apply for an NRC design certification, an early site permit, and a combined construction/operating license under 10 CFR Part 52 to reduce the overall time and expense for future license applications. Using the same certified design could allow construction of multiple units at a site without reopening or repeating the design review process.

The following are recommendations for actions that will enhance the feasibility of placing SMRs at remote sites within the United States:

- Review the current regulatory process, including the general design criteria, to identify requirements that apply only to LWR designs. In addition, identify new SMR-specific requirements that may be necessary to ensure that an equivalent level of safety is maintained. Incorporation of the results from these two efforts into a set of requirements applicable to SMR designs would remove unnecessary burden and make the SMR licensing process more efficient.

- For the gas-cooled SMRs, develop fuel quality assurance requirements and testing and acceptance criteria for ceramic coated fuels. Review existing research and perform additional research to determine the probability of graphite fires occurring in advanced gas reactor designs.

- If a pressure-retaining containment is not included in the SMR design, demonstrate its fission product retention capability via a testing program that may include the use of a full-scale prototype plant. The testing should generate plant performance data sufficient to validate safety analysis tools over an extensive range of operating and accident conditions.

Indemnification of reactors under the Price-Anderson Act

The Price-Anderson Act provides a system of financial protection for persons who may be liable for and persons who may be injured by a nuclear incident at reactors licensed by the NRC or operated by the Department of Energy. Licensees authorized to operate nuclear reactors of less than 10 MWt capacity are required by NRC to have and maintain financial protection in amounts ranging from $1 million to $2.5 million, depending on their power levels. Financial protection requirements for power reactors authorized to operate above 10 MWt and below 100 MWe are established in accordance with a formula designed to take into account the population in reasonably sized area around the reactor. Under the formula, population is weighted roughly in inverse proportion to the square of the distance from the reactor site.

These smaller reactors are not subject to the retrospective pooling arrangements (annual deferred premium) required for large power plants over 100 MW. In total, indemnification for small reactors under NRC's regulatory scheme would be approximately half a billion dollars for liability for a nuclear incident. Consideration should be given to whether this amount of compensation would be sufficient for the type of small reactors under discussion in this report.

5 ECONOMIC PERFORMANCE MODELING FOR GENERIC SMALL MODULAR REACTORS

In order to be a viable option for power production in remote areas, SMRs need to provide competitively priced (as determined by busbar cost) electric power. This chapter presents the range of cost parameters used to develop a generic 50 MWe and 10 MWe SMR. "Busbar cost" is the cost required to generate a kilowatt-hour of electricity as measured at the plant busbar, i.e., the conducting boundary in the plant where the generated electricity is transferred to the external electrical grid.

5.1 Modeling Parameters

As discussed in Chapter 2 of this report, SMRs were built and operated by the United States Army as early as the 1950s. The successful operation of the SM-1, at Fort Belvoir, Virginia, and MH-1A, which was used as substitute power in remote areas, helped lead the way to the building of small commercial power plants in the 1960s and 1970s. One of the first small plants built was Peach Bottom 1 in 1967. Peach Bottom was a 46 MWe gas-cooled reactor built by General Atomics (GA) located in Pennsylvania. The total capital cost for the plant was $232 per kilowatt-hour (kWe). A second gas-cooled reactor, Fort St. Vrain (343 MWe), was put in service in 1979 in Colorado and was also built by GA. Fort St. Vrain had a capital cost of $308 per kWe. Both of reactors used steam turbines to generate electricity, but an expected feature in advanced gas SMRs will be the use of direct-drive gas turbines.

Although there is an extensive history with the building and operating of relatively small reactors, many of the new designs and concepts discussed in the previous chapters are still in an early stage of development. For example, some power plant vendors have the basic layout of the facility, perhaps even down to some of the smallest details, but design tradeoffs and optimizations have not been done, and the detailed engineering to build the plant has not been completed. As a result, some of the cost information received from the February announcement was sparse. However, many of the features that are being incorporated in SMR designs are either being used in current larger plants or being tested for near-term operation. One such design is being developed by ESKOM, a utility in South Africa, which is proposing to start the construction of a 110 MWe pebble bed modular gas-cooled reactor (PBMR). Although this reactor is out of the size range of this study, it is a direct cycle helium-cooled concept like the SMR concept for the RS-MHR.

Based on the past experience with the building and operating of small reactors, a review of new design features being proposed by the various designers of SMRs, and vendors' estimated costs, and other parameters, the Department devised a method to estimate input costs for generic 50 MWe and 10 MWe SMRs, as shown in Table 2, values that in some cases differed from vendor estimates. For example, the economic levelization life used was reduced from a typical 30-year value, used by some vendors, to a 20-year time frame to ensure a standard basis for comparison.

Table 2
Cost Information for Generic 50 MWe and 10 MWe SMRs
(Year 2000 Dollars)

ITEM	Minimum 50 MWe	Minimum 10 MWe	Maximum 50 MWe	Maximum 10 MWe
Unit Capital Cost, $/kWe	1,950	3,950	5,067	11,330
Levelization Period, Yrs	20	20	20	20
Constant $ Fixed Charge Rate, %	11.2	11.2	11.2	11.2
Levelized Capital Cost, M$/Yr	10.9	4.4	28.3	12.6
O&M Costs, M$/Yr	5.5	2.6	9.4	5.6
Fuel Costs, M$/Yr	3.7	0.7	4.2	0.8

5.2 Busbar Costs for Generic Small Modular Reactors

The parameter values presented in Table 2 were used to develop maximum and minimum busbar costs for a 50 MWe and 10 MWe facilities (Tables 3 and 4). Note that the estimated busbar costs for the generic SMRs are for the "Nth of a kind" plant, typically the fifth plant of a given plant design constructed. The "Nth of a kind" construction cost are less than the construction cost for the "1st of a kind" plant, in that many of the engineering and design activities do not have to be repeated, and lessons learned from constructing the earlier plants are taken into account.

Table 3
Estimated Minimum Cost of Electricity,
(cents/kWh, Year 2000 $s)

Capacity	50 MWe	10 MWe
Capital	2.9	5.9
O&M	1.5	3.5
Fuel	1.0	1.0
Total	5.4	10.4

Table 4
Estimated Maximum Cost of Electricity,
(cents/kWh, Year 2000 $s)

Capacity	50 MWe	10 MWe
Capital	7.2	16.1
O&M	2.4	7.2
Fuel	1.1	1.1
Total	10.7	24.3

5.3 Estimated Cost of Electricity for Small Modular Reactors

Table 3 shows that the estimated minimum values for busbar costs vary from 5.4 cents per kWh at the 50 MWe size to 10.4 cents per kWh at the 10 MWe size. The maximum busbar cost, as shown in Table 4, varies from 10.7 cents per kWh at the 50 MWe size to 24.3 cents per kWh at the 10 MWe size.

5.4 Competitive Costs of Electricity in Remote Areas

Table 5 presents the delivered costs of electricity charged by certain utilities serving both the larger Alaskan cities (Anchorage, Fairbanks, and Juneau) as well as by those serving less populated Alaskan areas. Also, some data comparing the delivered costs of electricity to locations in Hawaii are included for comparison. These two states represent areas where nuclear-generated electricity from SMRs should be most applicable. While rates for residential customers as well as the lower rates charged industrial users are included, the industrial rates were chosen for comparison because they contain competitively low charges for transmission and distribution; thus, they approximate busbar cost. Note that many of these rates would be higher today (in year 2001), accounting for the recent increases in fossil fuel prices.

The industrial rate for electricity charged by selected Alaskan utilities shown in Table 5 varies from 5.9 to 36.0 cents per kWh. In Hawaii, the industrial rate charged, varies from 8.8 to 15.1 cents per kWh. The industrial rate for the more populated locations in Alaska and Hawaii (number of total customers >30,000) varies from 5.9 to 15.1 cents per kWh.

Table 5
Cost of Electricity to Users in Selected Locations
(1999 Financial & Production Data[3])

State	Utility	Location	Cost cents/kWh (Industrial or Commercial)	Cost cents/kWh (Residential)	Total Number of Customers	Net Generation MWh	Purchased Power MWh	Peak Power MWe
Alaska	Golden Valley Electric Assoc.	Fairbank	5.9	9.3	35,945	611,227	438,528	175
	Chugach Electric Assoc.	Anchorage	6.5	10.3	68,862	2,091,897	232,789	412
	Alaska Electric Light & Power	Juneau	7.2	9.6	14,623	66,533	246,896	59
	Sitka Municipal Utilities	Sitka	7.6	9.3	4,533	94,583	0	18
	Ketchikan Public Utilities	Ketchikan	7.7	9.5	7,090	90,863	68,135	28
	Anchorage Municipal L&P	Anchorage	7.8	9.9	29,567	704,704	215,090	151
	Nome Joint Utility Systems	Nome	13.4	20.8	1,781	29,298	0	5
	Bethel Utilities Corp	Anchorage	18.5	23.7	2,219	37,152	0	7
	Alaska Village Electric Coop Inc.	Anchorage	36.0	45.0	6,371	53,940	0	12
Hawaii	Hawaiian Electric Co.	Honolulu	8.8	13.0	273,968	4,391,007	2,965,718	1161
	Maui Electric Co.	Kahului	12.7	14.5	55,786	1,062,099	69,973	180
	Hawaii Electric Light Co.*	Hilo	15.1	20.6	59,744	599,875	380,603	167

*1997 data for Hawaii Electric Light Co.

Note: Costs have been escalated to year 2000 dollars.

5.5 Competitiveness of Small Modular Reactors

Using the generic estimates, the projected minimum values for busbar cost of 5.4 cents per kWh of electricity at the 50 MWe size is competitive with the current cost of electricity generation for industrial customers at all of the selected locations in Alaska and Hawaii, as shown in Table 5. Likewise, the projected minimum cost of electricity based on a 10 MWe SMR, 10.4 cents per kWh, is competitive with the current cost of electricity sold by five of the 12 selected utilities in Alaska and Hawaii.

The projected maximum values for busbar cost of 10.7 cents per kWh of electricity at the 50 MWe size is competitive with the cost of electricity sold by five of the 12 selected utilities. The generic 10 MWe SMR is estimated to be able to produce electricity at 24.3 cents per kWh and would be competitive with only one of the selected utilities shown in Table 5.

Nuclear-generated electricity costs from specific SMR designs may be better or worse than that projected by the model for the generic design SMRs. However, we believe the model reasonably brackets the range of costs that can be expected from SMRs of the 10 MWe and 50 MWe size.

6 CONCLUSIONS

Overall, it appears that there are SMR designs and concepts that meet the criteria set forth in Senate Report 106-395. These new small reactors have no insurmountable technical issues to hinder development and deployment, and the projected range of costs for SMRs are comparable with current rates charged in some remote communities. Moreover, U.S. experience with small reactors has shown that these facilities can be safely constructed and operated. One issue requiring further study is the lack of supporting infrastructures for supplying fuel for each of the SMRs. Depending on the fuel type, suitable fuel fabrication facilities may not currently exist, and would need to be constructed and qualified. Currently, this may be of particular concern for gas-cooled reactors using graphite fuel, although potential pebble-bed reactor development might alter this situation. Further, some of the new SMRs are cooled by gas or liquid metals, and some of the more viable designs and concepts face licensing questions which are outside the traditional NRC light water reactor experience. As a result, the current regulatory guidelines may not be fully applicable. Additionally, it is not clear to what extent the elimination of a conventional containment would be acceptable to the NRC. This would need to be explored. Also, it would be beneficial to further refine the SMR cost estimates after determining the full extent of potential regulatory issues in order to make fully informed decisions regarding development and deployment in remote communities.

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1. "Design and Development Status of Small and Medium Reactor Systems," IAEA-TECDOC-881, 1995.

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3. *Electrical World*, 2001 Edition.

APPENDIX: LIGHT WATER REACTOR REGULATORY FUNDAMENTALS

This appendix provides a brief technical description of the current regulatory structure that has been applied to light water reactors (LWR). It includes a brief history of the current regulations and some of the NRC's initiatives to implement risk-based techniques.

A.1 Cornerstones of Safety

The U.S. Nuclear Regulatory Commission (NRC) has the responsibility of ensuring adequate protection of the public health and safety, the common defense and security, and the environment in the use of nuclear materials in the United States. The NRC's current regulatory framework for reactor oversight is shown in Figure 2. It is a risk-informed, tiered approach to ensuring plant safety. There are three key strategic performance areas:

- Reactor safety (avoiding accidents and reducing the consequences of accidents if they occur);
- Radiation safety for both plant workers and the public during routine operations; and
- Protection of the plant against sabotage or other security threats.

Within each strategic performance area are principles (cornerstones) that reflect the essential safety aspects of facility operation. Satisfactory licensee performance in these areas provides reasonable assurance of safe facility operation. The NRC cornerstones are described below (see Figure 2):

Initiating Events - Focuses on operations and events at a nuclear plant that could lead to a possible accident if plant safety systems did not intervene. These events could include equipment failures leading to a plant shutdown, shutdowns with unexpected complications, or large changes in the plant's power output.

Mitigating Systems - Measures the function of safety systems designed to prevent an accident or reduce the consequences of a possible accident.

Barrier Integrity - For light water reactors (LWR), there are three important barriers between the highly radioactive materials in fuel within the reactor and the public and the environment outside the plant. These barriers are the sealed fuel rods, the heavy steel reactor vessel and associated piping, and the reinforced concrete containment building surrounding the reactor.

Emergency Preparedness - Measures the effectiveness of the plant staff in carrying out its emergency plans.

Occupational Radiation Safety - Monitors the effectiveness of the plant's program to control and minimize dose to the plant workers.

Public Radiation Safety - Measures the procedures and systems designed to minimize radioactive releases from a nuclear plant during normal operations and to keep those releases within federal limits.

Physical Protection - Requires security personnel and a variety of protective systems to guard vital plant equipment.

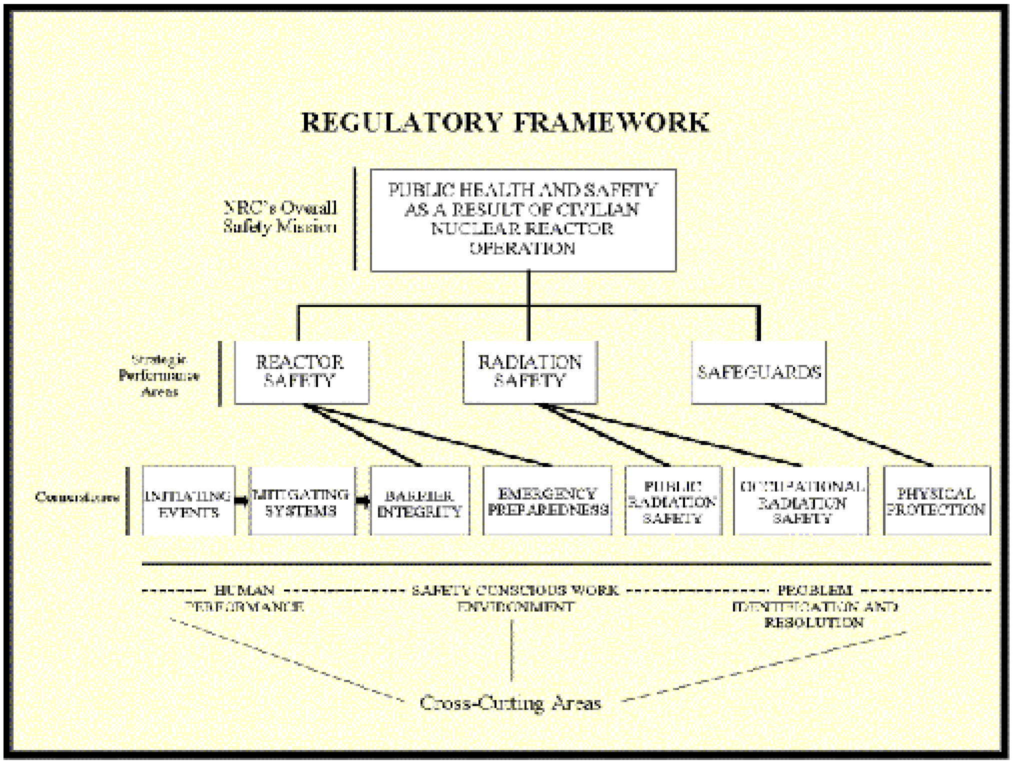

Figure 2. NRC Cornerstones of Safety

A.2 Defense-in-depth

Defense-in-depth is a philosophy that uses various layers of requirements that help ensure safety is achieved through multiple, diverse, and complementary means. These layers can include 1) conservative plant design, 2) high safety system reliability, redundancy and/or diversity, 3) mitigation of fission product releases in the event that one or more barriers fail during an accident, and 4) emergency planning to protect the public should an accident lead to a radioactive release to that exceeds allowable limits.

The defense-in-depth philosophy has traditionally been applied in reactor design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive materials.

It has been and continues to be an effective way to account for uncertainties in equipment and human performance. If a comprehensive risk analysis is done, it can be used to help determine the appropriate extent of defense-in-depth (i.e., the balance among core damage prevention, containment failure, and consequence mitigation) to ensure protection of public health and safety.

Consistency with the defense-in-depth philosophy is maintained if:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (i.e., no risk outliers).
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The General Design Criteria in Appendix A to 10 CFR Part 50 is maintained.

A.3 Risk-Informed Techniques

In 1991, the Commission instructed NRC staff to investigate the feasibility of using more performance-based regulations that focused on a "result to be obtained, rather than prescribing to the licensee how the objective is to be obtained." Probabilistic Risk Assessment (PRA) is a methodology that can be used to provide a structured analytical process to assess the likelihood and consequences of severe LWR reactor accidents. The NRC's first application of PRA was the Reactor Safety Study, WASH-1400, in 1975. Since that time the NRC has made use of risk assessment to address complex safety issues such as those involved in the regulations addressing Station Blackout (SBO), Anticipated Transient Without Scram (ATWS), and Pressurized Thermal Shock (PTS). Risk assessment is used to determine the importance of generic safety issues and in the preparation and evaluation of responses to various generic letters. There are a number of ongoing NRC activities in which PRA has been and will continue to be applied. Most notably, these activities include:

- Evaluation of operational events to provide lessons learned from plant experience data (i.e., using PRA techniques, such as the Accident Sequence Precursor methodology, to estimate the likelihood that a given event initiated at a particular plant would progress to a severe accident);
- Review of PRA applications submitted in support of standard design certification for a nuclear power plant and plant licensing (i.e., for the certification and licensing of advanced light water reactors designs and plants); setting the priority of NRC staff activities, such as inspections, review of potential generic safety issues, and research (i.e., the staff uses PRA results to focus its licensee monitoring activities on risk-significant plant components and systems, as well as to prioritize generic safety issues according to their importance to risk);

- Review of PRA applications submitted by licensees in support of proposed license amendments (i.e., for evaluation of proposed changes to plant technical specifications); and
- Performance of regulatory analyses (i.e., to determine whether potential changes considered for resolution of certain generic safety issues reduce risk sufficiently to justify their cost).

In an effort to improve plant safety and reduce unnecessary burden through the most effective use of PRA, interactions are continuing among the NRC, licensees, industry organizations, and public interest groups. The expected results from these activities are a better focus by the NRC and by licensees on those licensing actions and regulatory practices which have a significant impact on plant risk and less emphasis on those which have less impact on plant risk.