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Generation IV Roadmap

Description of Candidate Water-Cooled Reactor Systems Report

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MEMBERS OF THE WATER-COOLED REACTOR SYSTEMS TECHNICAL WORKING GROUP

John C. (Jack) Devine, Jr.	Co-chair	Polestar, United States
Kenneth R. Hedges	Co-chair	Atomic Energy of Canada Limited, Canada
Philip E. MacDonald	Technical Director	Idaho National Engineering and Environmental Laboratory, United States
Mario Carelli		Westinghouse Electric Company, United States
John Cleveland		International Atomic Energy Agency, United Nations
Michael L. Corradini		University of Wisconsin, United States
Dario Delmastro		Comisión Nacional de Energía Atómica, Argentina
David J. Diamond		Brookhaven National Laboratory, United States
Kazuyoshi Kataoka		Toshiba Corporation, Japan
Philippe Lauret		Framatome – ANP, France
Yoon Young Lee		Doosan Heavy Industries Company, Korea
Yoshiaki Oka		University of Tokyo, Japan
Akira Omoto		Tokyo Electric Power Company, Japan
Jong Kyun Park		Korean Atomic Energy Research Institute, Korea
Noval A. Smith, Jr.		Dominion Virginia Power Company, United States
Antonio Teixeira e Silva		IPEN/CNEN, Brazil
Alfredo Vasile		Commissariat à l'Energie Atomique, France
Gary S. Was		University of Michigan, United States

OTHER CONTRIBUTORS

Bobby Abrams	GRNS Representative	Duke Engineering, United States
Todd Allen	RIT Representative	Argonne National Laboratory, United States
Ralph Bennett	RIT Representative	Idaho National Engineering and Environmental Laboratory, United States
Jacopo Buongiorno		Idaho National Engineering and Environmental Laboratory, United States
Richard Schultz		Idaho National Engineering and Environmental Laboratory, United States

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EXECUTIVE SUMMARY

This Technical Working Group was charged with identifying and evaluating advanced water-cooled reactor nuclear energy system concepts under the Generation IV Program. The initial activity, as described in this report, is the assessment and screening of candidate concepts for potential Generation IV participation. The subsequent technical working group evaluation will support the selection of concepts and technology for research and development (R&D) support.

Advanced water-cooled-reactor nuclear energy system concepts were identified by the technical working group and via a formal DOE "Request for Information" issued in April 2001. A total of 38 nuclear energy system concepts covering a wide range of design features, both evolutionary and innovative in nature, were received. Some of these were similar (and in some cases nearly identical) while others were unique.

To establish a basis for meaningful and manageable comparison, all but one of the 38 concepts were consolidated into distinct ten sets, each of which includes a set of concepts with common key features or characteristics. These are (with their respective acronyms):

1. *Integral Primary System Reactors (IPSRs).* These light water reactor concepts are characterized by a primary system that is fully integrated in a single vessel, which makes the nuclear island more compact and eliminates the possibility of large releases of primary coolant. The primary-coolant mode of circulation is either forced or natural. All the proposed concepts are thermal reactors and make use of low-enrichment-uranium oxide or conventional mixed uranium-plutonium oxide (MOX)-fuel, clad with Zircaloy.

2. *Loop Pressurized Water Reactors (Loop PWRs).* These are modified loop-type pressurized water reactors (PWRs) with a water-filled safeguard vessel (or a series of vessels and pipes) enveloping the whole primary system.

3. *Simplified Boiling Water Reactors (SBWRs).* These are various size boiling water reactors (BWRs) with natural circulation in the core region, no re-circulation pumps, and, in most cases, highly passive decay heat removal systems.

4. *Pressure-Tube Reactors (PTRs).* These are Canadian deuterium-uranium (CANDU)-type reactors with light water cooling and fuel that is slightly enriched. Various thorium fuel cycles have also been proposed. One concept features higher temperature and pressure conditions to increase the thermal efficiency. The focus of the next generation CANDU reactor (NG CANDU) is on significantly reducing capital costs.

5. *Supercritical Water-Cooled Reactors (SCWRs).* These are a class of high-temperature, high-thermal-efficiency water-cooled reactors with a primary coolant system that operates above the thermodynamic critical point of water (374.1°C, 221.2 bar). The core may have a thermal or fast neutron spectrum depending upon the specific design, and both light water and heavy water moderation have been proposed. Plant efficiencies between 40 and 45% can be obtained with the use of supercritical water.

6. *High-Conversion Water-Cooled Reactors (HCRs).* These are various reduced-moderation reactor cores designed to use uranium more efficiently (conversion ratio near 1.0) and minimize the reactivity swing. Both light and heavy water, either boiling or pressurized, are proposed as coolant. The positive void coefficient is reduced by the use of neutron streaming assemblies and pancake-type cores.

7. *Pebble Fuel Reactors (PFRs). The* principal thrust of these concepts is the use of a fluidized bed of ceramic or metallic fuel pebbles in sizes ranging from a few mm up to about 10 mm, which keeps the *fuel at low temperatures, enabling higher core power densities and safer operation.*

8. *Advanced Light Water Reactors with Thorium/Uranium Fuel (Thorium Fuel).* These are advanced light water reactors (ALWRs) with either homogeneously mixed thoria-urania fuels or various seed and blanket arrangements using both oxide and metal fuel. These fuels are designed to provide a variety of ALWRs with better resource utilization and more proliferation resistance.

9. *Advanced Water-Cooled Reactors with Dry Recycling of Spent LWR Fuel (Dry Recycle).* This fuel cycle consists of an oxidation/reduction process to recycle spent light water reactor (LWR) fuel into CANDU reactors or, with added enrichment, back into ALWRs. The dry recycle process prevents the separation of most of the fission products from the plutonium, thereby making the plutonium un-useable in a nuclear weapon.

10. *Advanced Light Water Reactors with Plutonium and Minor Actinide Multi-Recycling.* These are ALWRs with either normal moderation or reduced moderation cores that burn plutonium and minor actinides. Multi-recycling of the plutonium and minor actinides has the potential to reduce the high-level waste burdens, extend uranium resources, reduce enrichment requirements, and therefore, improve the sustainability of nuclear power.

Sections 2.1 through 2.10 of this report are summary descriptions of the nine sets. Much more detailed descriptions are provided in Appendices W1 through W10. The appendices also contain a preliminary evaluation of each concept set potential for meeting the Generation IV objectives, a score-sheet summary of this evaluation, the identification of the main R&D needs for each concept set, an estimate of the required time for deployment, and a statement of the technical working group judgment regarding the overall potential of the concept set. Appendix W11 provides an assessment of the one concept, the U-Np-Pu fuel cycle, that did not fit into any of the other concept sets.

A preliminary assessment of the key R&D needs resulted in the items listed below:

- Development of fuel cladding and structural materials with higher fast fluence and/or longer burnup limits (e.g., >50MWd/kg)

- Development of fuel cladding and structural materials for supercritical water-cooled reactor applications (e.g. pressures >221.2 bars, temperatures >374.1°C).

- Development of reliable and low-maintenance components for integral reactors and/or long-irradiation fuel cycles (e.g., in-vessel control rod drives, steam generators, pumps, pressurizers)

- Optimized core designs and fuel cycles for high conversion reactors

- Experimental verification of the performance of the simplified safety systems

- Quantitative evaluation of the economic and safety advantages and disadvantages of small-to-medium power modular systems vs. large-power monolithic systems

- Updating and validation of existing neutronic and thermal-hydraulic models, databases and predictive tools, e.g., neutron cross sections for high-conversion reactors, heat transfer correlations for supercritical water-cooled reactor designs, etc.

Many of these R&D needs apply to more than one of the concept sets. Table 8 of this report provides a correlation of R&D needs and water-cooled concept sets. It is the technical working group members' view that these R&D needs should be given consideration in the final determinations regarding Generation IV funding support.

The primary purpose of this phase of the Generation IV Program is to develop a full understanding of the candidate reactor systems or technologies and to conduct an initial "screening for potential" with respect to Generation IV goals. To that end, this report includes the technical working group recommendations for continued inclusion in the Generation IV evaluation. As presented in Section 3, the technical working group proposes to retain all ten of the concept sets for further assessment in the second phase of the Generation IV Roadmap. The technical working group proposes to eliminate from further consideration the individual Concept W15, the U-Np-Pu cycle concept. The rationale for this technical working group recommendation is provided in Appendix W11.

In summary, the technical working group evaluations and conclusions presented in this report provide a comprehensive and sound basis for subsequent screening, comparison with other (e.g., gas-cooled, liquid metal and "nonclassical") concepts, and selection of final R&D work to be supported under Generation IV.

Description of Candidate Water-Cooled Reactor Systems

1. INTRODUCTION

The overall goal of the Generation IV Program is to identify and develop next-generation nuclear energy systems that can be deployed over the next 30 years to help meet the world's energy needs throughout the 21st century. These next-generation energy systems are expected to offer significant advances in fuel cycle sustainability, along with improvements in safety, performance, and cost of energy in comparison with current plants.

Within the Generation IV Program, this Technical Working Group was charged to identify and evaluate advanced water-cooled-reactor nuclear energy system concepts. The initial activity, described in this report, was to assess and screen for potential candidate systems in order to establish a sound basis for subsequent additional evaluations, comparisons with other (nonwater) reactor concepts, and final selection of concepts and technology for research and development (R&D) support.

Advanced water-cooled-reactor nuclear energy system concepts were identified in a formal DOE Request for Information (RFI) issued in April 2001 to industry, national laboratories, academia, and international groups. This process resulted in submittal of 30 advanced water-cooled-reactor nuclear energy system concepts[a] by researchers and industry experts in Argentina, Brazil, Canada, Italy, Japan, Korea, and the United States. In addition, the technical working group itself collected information on eight concepts, yielding a total of 38 concepts for evaluation.

The technical working group consolidated all but one of the 38 reactor and fuel cycle concepts into ten distinct concept sets, based on their key common features. The technical working group then conducted a comprehensive evaluation of these ten sets in order to determine their potential to achieve the Generation IV goals. This evaluation was used as a foundation for the initial screening step (Screening for Potential) in which any candidates (either individual concepts or concept sets) determined to have inadequate potential for a subsequent Technical Working Group recommendation for Generation IV Program R&D support were eliminated from further consideration.

The evaluation methods and conclusions are described in subsequent sections of this report and in the report appendices. The report is organized as follows. Section 2 presents a summary description of the concept sets. Section 3 reports a summary evaluation of the concept sets. Appendices A through J present detailed descriptions and evaluations of each concept set, with score sheet summaries of each evaluation, identification of the main R&D needs for each concept set, an estimate of the required time for deployment, and an initial technical working group judgment regarding their Generation IV potential. Each appendix is organized in a similar manner, with Section 2 of each appendix containing a relatively complete description of the concepts, described by the concept submitter. The descriptions were adapted from materials provided by the concept developers and may not necessarily reflect the judgment of the technical working group, which is reported, instead, in Section 3 of each appendix, Potential for Concept Meeting Generation IV Goals.

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a. Not surprisingly, there was a great deal of variation in the scope, depth, and completeness of the responses. Some respondents provided numerous supplemental papers and documents, but many did not provide any additional information. Some respondents made clear the intended fuel cycle technologies, and others did not. There were also a number of "partial concepts" submitted, primarily fuel cycle concepts that could fit into a wide variety of reactor types. We are assuming for the purposes of the Generation IV Roadmap that the various fuel cycle concepts can be used in a typical ALWR.

2. CONCEPT DESCRIPTIONS

As mentioned above, the technical working group consolidated all but one of the initial 38 water-cooled reactor concepts and fuel cycles into ten concept sets, based on central characteristics and features:

1. Integral primary system reactors (Appendix W1)
2. Loop PWRs (Appendix W2)
3. Simplified BWRs (Appendix W3)
4. Pressure-tube reactors (Appendix W4)
5. Supercritical water reactors (Appendix W5)
6. High-conversion water-cooled reactors (Appendix W6)
7. Pebble fuel reactors (Appendix W7)
8. ALWRs with thorium fuel (Appendix W8)
9. Water-cooled reactors with dry recycling fuel (Appendix W9)
10. ALWRS with plutonium and minor actinide multi-recycling (Appendix W10).

One individual concept, the U-Np-Pu fuel cycle did not fit well into any of the concept sets and was evaluated by itself. The results of that assessment can be found in Appendix W11 of this report.

To help understand the concepts, organize its thinking, and identify a manageable number of concept sets, the technical working group constructed a large table (referred to as the master table). The master table contains the following information for each individual concept: reactor size, plant design approach, coolant and moderator and their physical state, cycle, thermal efficiency, reactivity control, primary system layout and mode of circulation, neutron spectrum, fuel form, cladding materials, irradiation cycle and refueling, decay heat removal system, containment, important safety characteristics, proliferation characteristics, resource utilization, economic characteristics, and R&D needs. Appendix W12 of this report presents the master table, in two forms: a 40-page segmented version and a one-page version with fine print that can be seen by zooming in on it. From an inspection of this table and discussion and review of the characteristics of the various concepts, the technical working group placed the 38 individual concepts into the ten concept sets listed above.

The individual concepts that were submitted to the DOE in response to their Request for Information are labeled *W1, W2, etc*., in chronological order of receipt. The concepts described by the technical working group members were labeled *TWG1, TWG2, etc*. As mentioned above, these individual concepts are summarized in Appendix W12. The concept sets are summarized in Appendices W1 through W10. The use of W1, W2, etc., to describe both the individual concepts submitted to the Request for Information and the concept set appendices may be confusing to some readers. However, wherever we refer to a concept set description we refer to *Appendix WX*, and when we refer to an individual concept we say *Concept WX*.

The following sections summarize the nine concept sets. A complete explanation of each concept set is provided in its appendix.

2.1 Integral Primary System Reactors

Over the past several years, Integral Primary-System Reactor (IPSR) concepts have gained considerable interest within the United States, and internationally, as testified by the number and origin of the proposed concepts that fell into this class (see Table 1), i.e., a total of seven reactors, three from the United States, two from Japan, one from Korea, and one from Argentina. The best known of the recent concepts under development is probably the IRIS reactor (International Reactor Innovative & Secure), initiated by the Westinghouse Electric Co., Massachusetts Institute of Technology, and University of California at Berkeley through a DOE Nuclear Energy Research Initiative (NERI) grant and currently being developed by a collaboration of about 18 research and industrial partners in nine countries. IRIS is one of the four reactor concepts currently being evaluated by the NRC for early deployment in the United States. CAREM, a project of the Argentina's Commission Nacional de Energía Atómica (CNEA) was initiated over 15 years ago and was used as a reference design in a recent joint-study performed by the International Atomic Energy Agency of the United Nations (IAEA), the Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) and OECD's International Energy Agency (IEA). Several experimental facilities have been constructed to test various aspects of the CAREM concept. The SMART design being developed in Korea is also widely known and has been the subject of various international studies. Korea has recently announced that a prototype of the SMART reactor will be built starting in 2002. The Multi-Application Small Light Water Reactor (MASLWR) is also being studied through a NERI grant at the Idaho National Engineering and Environmental Laboratory (INEEL), University of Oregon and Bechtel Power Corporation.

Table 1. Summary of integral primary-system concepts submitted to DOE for the Generation-IV Program.

Gen-IV Designation	Proposer	Size	Coolant State/Pressure	Mode of Circulation[a]	Containment
W10 (SMART)	Chang (KAERI, South Korea)	330 MWth	Pressurized, 15.0 MPa	Forced	Spherical guard vessel with suppression pool plus traditional containment
W14 (CAREM)	Beatriz-Ramilo (CNEA, Argentina)	100–150 MWe	Pressurized, 13.0 MPa	Natural	With suppression pool
W16 (PSRD)	Ishida (JAERI, Japan)	100 MWth	Pressurized, 3.0 MPa	Natural	Partially filled with water
W17 (MRX, Ship Propulsion)	Ishida (JAERI, Japan)	100 MWth	Pressurized, 12.0 MPa	Forced	Completely filled with water
W18 (IRIS)	Carelli (Westinghouse, USA)	100–350 MWe	Pressurized, 15.5 MPa	Forced	HP spherical with suppression pool
W25 ("Daisy")	Buongiorno (INEEL, USA)	50–150 MWe	Boiling, 7.4 MPa	Natural	HP spherical, dry
W26 (MASLWR)	Modro (INEEL, USA)	35 MWe	Pressurized, with some boiling, 10.5MPa	Natural	Partially filled with water

a. *Natural* indicates full natural circulation, no pumps. *Forced* relies mainly on pumped flow. However, even the forced circulation reactors have a significant degree of natural circulation.

The IPSR concepts maximize the use of existing LWR technology, which is engineered in innovative ways to improve safety and simplify the plant. The main characteristic of these reactors is the integration of the whole primary system within a single pressure vessel. Because a catastrophic failure of the vessel is considered to be incredible, this eliminates (by design) the most important postulated accident for current LWRs, the large release of primary coolant from the rupture of an external-loop pipe (a large loss-of-coolant accident or LOCA). More generally, these reactors are characterized by the adoption of the so-called "safety by design" approach, i.e., an attempt is made to eliminate or reduce the possibility of the main accident initiators by design rather than having to mitigate the consequences of those accidents. For example, integration of the primary system makes it easier to achieve a higher degree of natural circulation of the primary coolant, which makes loss-of-flow accidents benign. Similarly, the utilization of in-vessel control-rod drives eliminates the possibility of control-rod ejection accidents. Also, a number of the IPSR concepts use a high-pressure containment and/or various water-filled compartments to basically eliminate the consequences of small-to-medium LOCAs (which are historically the accidents yielding the worst consequences). The water inventory within the reactor pressure vessel after a LOCA is maintained by reducing the pressure differential between the vessel and containment, thus reducing the driving force across the rupture and ultimately the coolant loss.

Three subgroups can be identified within the IPSR reactor class:

1. Reactors with traditional pressurized water reactor (PWR) pressure and temperature operating conditions

2. Reactors with somewhat lower-pressure water coolant

3. Reactors with boiling water coolant.

A brief description of these three subgroups is presented in Subsections 2.1.1, 2.1.2, and 2.1.3 below, respectively. However, this categorization will not be used for evaluation of the potential for meeting the Generation-IV goals (see Section 3), i.e. the different IPSRs will be evaluated together.

2.1.1 PSRs with Traditional PWR Operating Conditions (W10, W14, W17, W18)

These are small- or medium-size PWRs (30–350 MWe) with the reactor pressure vessel housing the whole primary system, including the core and the core support structures, the steam generators, the pressurizer, and the pumps. The steam generators are located in the annulus between the core barrel and the reactor pressure vessel wall. Both straight- and helical-tube steam generators are being considered. A pressurizer with either active heaters and sprayers or passive control with or without nitrogen gas pressure is located in the reactor pressure vessel upper head. The control rods are inserted from the reactor pressure vessel top. Internal control rod drives will be used in some of the concepts. The smaller-size concepts rely on full-power natural circulation of the primary coolant, while the larger-size concepts make use of canned-motor pumps or fully internal spool pumps while maintaining a relatively large natural-to-forced-circulation flow ratio. The operating pressure ranges from 12 to 15 Mpa; the inlet and outlet temperatures range from about 270 to 330°C.

The core of these reactors is made of a modest number of PWR fuel assemblies with uranium oxide fuel and modified pitch and fuel rod diameter. Some concepts adopt a triangular lattice, and some allow for the use of MOX fuel. To maximize the irradiation cycle (up to 5 years) and to compensate for the loss of reactivity associated with the smaller-diameter core, the enrichment is slightly larger in most of the designs than in current LWRs (4 to 5% versus 3 to 4%). Most concepts adopt a single batch refueling strategy, with replacement of the entire core every 4 to 5 years, which reduces fuel handling as well as spent fuel storage requirements but yields lower burnups and slightly higher fuel costs than in equal-length conventional cycles with partial refueling. Note that the single-batch long irradiation cycle is a common, but not an essential characteristic of these systems, which can also be operated with a conventional multibatch refueling approach of intermediate length (i.e., 12–18 months). The use of diluted boric acid is eliminated in all concepts and long-term control of the core reactivity is performed mainly by means of the control rods and burnable poisons, e.g., gadolinium, erbium, and boron. Because of the boron elimination, some designs feature alternative means to control the reactivity during cold shutdown and refueling.

2.1.2 Small IPSRs with Low-Pressure Water Coolant (W16, W26)

These are small-size (<100 MWe) pressurized water reactors whose operating pressure and temperatures are reduced to improve safety (i.e., smaller accumulated energy, larger safety margins) and simplify the plant (i.e., reliance on fully passive emergency systems). Some coolant boiling is allowed in the Multi-Application Small Light Water Reactor (MASLWR) design. High capacity factors are pursued by increasing the irradiation cycle (up to 10 years) and by adopting full-power natural circulation for greater reliability. Because of the lower operating conditions, the thermal efficiency of these plants is relatively low (<30%).

2.1.3 IPSRs with Boiling Water Coolant (W25)

This reactor is basically a small-size (<150MWe) natural-circulation BWR with an indirect cycle and a fully passive decay-heat removal system. The reactor operating temperature and pressure are 290°C and 7.4 MPa, respectively. The steam generated in the core is condensed in condensing units located within the steam dome at the top of the pressure vessel. A key feature of this reactor is that the secondary water (flowing in the condensing unit tubes) is maintained liquid at a pressure higher than the primary system pressure (8.0 MPa) so that if a tube rupture occurs, there is no release of the primary coolant. Therefore, to generate steam, the flow of secondary water must be subjected to an abrupt and large pressure drop in a dedicated throttling valve that causes some water to flash to steam. This steam is dried in a moisture separator and then is sent to the turbine. From this point on, the power cycle is similar to that of traditional LWRs. Thermal efficiencies up to 29% are possible, somewhat smaller than typical LWRs because of the large pressure drop in the throttling valve.

2.2 Loop Pressurized Water Reactors

This set comprises two reactor concepts. The first is the Simple & Intelligent PWR with Bloc Type/Double Vessel Utilizing Compact Thoria-Urania Dispersed Metal Fuel (Bloc). The other is the Multipurpose Advanced Reactor, Inherently Safe (MARS). The common innovative characteristic of these reactors is use of a safeguard vessel (or series of vessels and pipes) that envelopes the whole primary system (i.e., the main pressure vessel, steam generators, control rod drives, and pressurizer) for mitigation of primary system component failure. However, significant differences exist. The general characteristics of these two reactors are compared in Table 2.

The Bloc reactor is a large pressurized water reactor (PWR) with an electrical output >1,500 MWe whereas MARS is a small PWR (150MWe). The Bloc PWR operates at typical PWR pressures and temperatures, while MARS operates at substantially lower temperatures and pressures for reduction of the structural materials oxidation and reduction of the energy accumulated in the primary system. Some design features of the Bloc Type PWR are revolutionary compared to the reference ALWR. However, the concept builds on the Korean ALWR designated as the APR1400 (Advanced Power Reactor, 1400 MWe) that is currently in the final stage of development and is to be in commercial operation in 2010 in Korea.

The design features of the MARS reactor are more evolutionary. The MARS project started in 1983 with the objective of developing a reactor to be used for a wide range of applications, including desalination and district heating. The MARS design was developed over 15 years, and the proponents

Table 2. General characteristics of the Loop PWRs.

	Bloc-Type PWR	MARS
Gen-IV Designation	W11 (Bloc Type PWR)	W3 (MARS)
Proponent	Park (KAERI, Korea)	Sorabella (University of Rome, Italy)
Power (MWe)	>1500	150
Thermal Efficiency	35%	25%
Coolant/Pressure	Light water, pressurized, 15.0MPa	Light water, pressurized, 7.5MPa
Circulation Mode	Forced	Forced
Fuel	Thoria-Urania dispersed in Zr Metal	LEU oxide
Cycle Length	10 years	18 months
Decay Heat Removal	Passive (air on the outer containment surface)	Passive (LP emergency condensers)
Special Features	Safeguard vessel around the primary system	Double-walled primary system
Safety Features	LOCAs and severe accident mitigated	LOCAs and severe accident mitigated

claim it is almost ready for deployment after minor verification/validation of its engineering features. The MARS would be adequate for deployment in countries with a need for small-to-medium-size plants.

2.3 Simplified Boiling Water Reactors

The BWR designs, successfully promoted by the General Electric Co (GE) and their licensees, have been built from almost the beginning of the commercial nuclear era. The Generation II concepts, perhaps best represented by the BWR-6, have been eclipsed by the more technically advanced boiling water reactor (ABWR) design—a Generation III plant. Because of the established record of success achieved by the BWR designs, there is every reason to believe that there will be commercially successful Generation IV SBWR designs. The designs submitted for consideration are summarized in Table 3. Of the five designs, there is one monolithic design submitted by GE, three modular designs (two from the United States and one from Japan), and one special purpose concept designed to desalinate water (from Japan).

Table 3. Summary of simplified boiling water reactor concepts submitted to DOE for the Generation-IV Program.

Gen-IV Designation	Proposer	Size	Coolant State/Pressure	Containment
W7 (SMART)	Khatib-Rahbar (Energy Research, Inc, USA)	50–300 MWe	Boiling	Large volume BWR/PWR hybrid
W8 (SBWR-Purdue)	Ishii (Purdue University, USA)	50 MWe	Boiling; 7.2 MPa	Small
W23 (LSBWR)	Heki (Toshiba, Japan)	300 MWe	Boiling; 7.0 MPa	Smaller than conventional BWR (with suppression pool)
W13 (ESBWR)	Rao (General Electric, USA)	1380 MWe	Boiling	Large (with suppression pool)
W22 (Desalination)	Kataoka (Toshiba, Japan)	589 MWth	Boiling; 7.0 MPa	Small (with suppression pool)

The best known of the submitted concepts are the European Simplified BWR (ESBWR), submitted by GE (W13), and the SBWR design, submitted by Purdue University (W8)—since Purdue's design is based substantially on the original GE SBWR design that was submitted as a licensing candidate a few years ago. The U.S. Nuclear Regulatory Commission did not grant a license to the GE's SBWR design, since GE withdrew it from consideration before the process was completed.

Significant common features of the group are as follows:

a. These BWRs are all direct cycle light water reactors with conventional energy conversion systems and efficiencies (with the exception of the desalination plant, W22).

b. All rely on natural circulation, rather than on mechanical or jet pumps, either internal or in recirculation loops.

c. All utilize passive safety features similar to those used in the reference plant (ABWR).

d. All but one of the concepts use relatively conventional uranium oxide, Zircaloy clad fuel. The SBWR-Purdue, Concept W8, expressed a preference for 5% enriched ThO2-UO2 fuel. However, the backup fuel for this concept is low-enrichment uranium (LEU).

e. The remaining SBWR power reactors, although specifying low enrichment uranium as their chosen fuel, do mention backup fuels, which are ThO2-UO2 (SMART), medium-enriched UO2 for very high burnup (LSBWR), and mixed uranium-plutonium oxide (MOX) rods (ESBWR).

f. All the modular concepts feature long fuel cycles ranging from 10 years (SBWR and SMART, W8 and W7) to over 15 years (LSBWR, W23). Due to its 15-year fuel cycle, the LSBWR design does not include a spent fuel pool. The ESBWR concept (W13) features intermediate length fuel cycles. Refueling must be accomplished with the system offline.

g. The modular concepts are designed, to one degree or another, for a major portion of the system construction to be performed in a factory. The factory-produced system is then transported and deployed at the site. Examples of this approach are SMART (W7) and SBWR (W8). Although not clear in the concept description, portions of the LSBWR concept (W23) seem to be factory constructed.

h. The containments fall into two general categories: large volume—BWR/PWR hybrid (SMART, W7) and volumes of various sizes with suppression pools (W8, W13, W22, and W23).

The concepts differ in size and structural approach, covering both modular and monolithic designs with power ratings from 50 to 1380 MWe. They also differ significantly in safety system design, in plant layout and equipment configurations, in containment design, in operating characteristics, and in level of design maturity (some are highly conceptual, while others are well developed).

The SBWRs can be divided into three subgroups:

1. Monolithic SBWRs
2. Modular SBWRs
3. Special-purpose.

A brief description of these three subgroups is presented in subsections 2.3.1, 2.3.2, and 2.3.3 below, respectively.

2.3.1 Monolithic SBWR: (W13)

The ESBWR is a 4000-MWth (approximately 1400-MWe) boiling water reactor that uses the same basic passive technology and simplified design as its predecessor (the 2000-MWth SBWR). The system makes use of existing technology whenever possible—such as GE's fine motion control rod drive system. The ESBWR plant design relies on the use of natural circulation and passive safety features to enhance plant performance and simplify the design (such as reductions in the required numbers of control blades and control rod drives). Use of natural circulation has allowed elimination of several systems—such as the recirculation pumps. Adequate natural circulation behavior has been achieved using shorter fuel and an improved steam separator (to reduce the core pressure drop), and a 7-meter chimney to enhance the driving head.

The ESBWR uses isolation condensers for high-pressure inventory control and decay heat removal under isolated conditions. The isolation condenser system has four independent high-pressure loops, each containing a heat exchanger that condenses steam on the tube side. The tubes are in a large pool, outside the containment. The steam line connected to the vessel is normally open, and the condensate return line is normally closed.

In the event of an accident, the vessel is depressurized rapidly to allow multiple sources of safety and nonsafety systems to provide water makeup. By eliminating all large penetrations in the lower part of the reactor vessel, the ESBWR core will remain covered by water during any rapid depressurization event. Hence, the makeup system has only to provide a slow water makeup to account for loss of inventory resulting from boil-off by decay heat. The makeup water flows into the vessel by gravity, using the Gravity Driven Cooling System, instead of relying on pumps and their associated support systems. The ESBWR uses an automatic depressurization system to depressurize the vessel.

Containment heat removal is provided by the Passive Containment Cooling System, consisting of four safety-related low-pressure loops. Each loop consists of a heat exchanger open to the containment, a condensate drain line, and a vent discharge line submerged in the suppression pool. The four heat exchangers, similar in design to the isolation condensers, are located in cooling pools external to the containment.

2.3.2 Modular SBWRs: (W7, W8, & W23)

Modular SBWRs are small- or medium-size BWRs (50–300 MWe) designed to have major components manufactured in factories and then shipped in toto to the plant site. The degree to which each of these concepts will be completed in a factory and then shipped to the plant site differs from one to another—and was not well defined in the concept descriptions. The modular BWRs, as a group, increase proliferation resistance by tending to have long operating cycles.

2.3.3 Special Purpose SBWR: (W22)

Concept W22 is a coupling between a small natural circulation BWR and a reverse osmosis seawater desalination system through turbine-driven-pumps as an interface. Both the BWR and the reverse osmosis system are simple designs that improve the economics as well as the plant reliability. Use of turbine-driven pumps, which are often used in nuclear power plants, also enhances the economics as well as the safety because they can eliminate use of an extra heat exchanger as an interface between the nuclear system and the desalination system. All these technologies are well proven and existing, so that neither large R&D nor new investments in manufacturing facilities is necessary.

The core power density is decreased instead of changing the core and/or fuel designs. This decrease in power density results in simplification in the coolant circulation system of the BWR because the natural circulation cooling is sufficient for a core with such low power density. The low power density also lengthens the refueling intervals and consequently enhances the availability of the plant.

2.4 Pressure Tube Reactors

Several advanced pressure tube reactor design concepts have been proposed as Generation IV reactors (see Table 4). A common feature of these designs is the adoption of light water as the coolant. All of these concepts have the pressure tubes oriented horizontally in order to take advantage of on-line fuelling, and they employ an indirect steam cycle. They can all be considered as advances on the CANDU-type reactor design. The key differences in the proposed concepts are in the moderator/calandria design and the fuel design.

The primary drivers of the three concepts are different. The main driver for the advances in the next generation CANDU design is improved economics, achieved principally through a capital cost and construction schedule reduction. Key features that enable the improved economics are reduction in the heavy water inventory, an increase in thermal efficiency, a smaller core, and a design based on modular construction. The Passive Pressure Tube Reactor (Passive PTR) design is focused on passive safety, while the High Conversion PTR design is focused on fuel cycle optimization. Each of these concepts is summarized in turn in the following three subsections.

Table 4. Generation IV pressure tube reactor concepts.

Concept	Key Features	Sponsor
W6, Next Generation CANDU (NG CANDU)	Light-water coolant Heavy-water moderator in calandria Slightly enriched uranium fuel	AECL
W28, Passive Light-Water Pressure-Tube Reactor (Passive PTR)	Light-water coolant Option 1: No separate moderator - Gas-filled calandria and graphite reflector, CANDU-type fuel Option 2: Light-water moderator & graphite matrix fuel	MIT
W5, High Conversion Pressure Tube Light Water Reactor (High Conversion PTR)	Light-water coolant Light-water moderator Gas-filled calandria Thoria-urania fuel	Kyung Hee University

2.4.1 Next Generation CANDU (W6)

The next generation CANDU design is based on the standard CANDU design with horizontal pressure tubes fuelled on line, with short fuel bundles and surrounded by a low-temperature heavy water (D_2O) moderator. The CANDU design features include high neutron efficiency, ease of construction, and localization. An inherent safety feature of the design is a passive moderator/shield tank heat sink surrounding the pressure tube core. The major innovations in the next generation CANDU are:

1. A more compact core design
2. Replacement of the heavy water in the reactor coolant system with light water
3. Slightly enriched uranium oxide fuel in CANFLEX fuel bundles

4. Higher thermal efficiency
5. Enhanced passive safety systems
6. Improved performance through advanced operational and maintenance information systems.

2.4.2 Passive Pressure Tube Reactor (W28)

Two variants of the Passive PTR concept have been proposed by the Massachusetts Institute of Technology (MIT). Both designs are based on high-power (>1000 MWe) versions of the current CANDU reactor design. The key differences are the design of the calandria and fuel, and the elimination of the Emergency Core Cooling System.

The dry calandria version has no moderator on the outside of the fuel channels. The light water coolant provides the required moderation, and there is a solid graphite reflector inner liner to the calandria. Under normal operation, the calandria space is filled with a low-pressure gas in balance with a water column in the containment building. In the event of a loss-of-coolant accident, the calandria is flooded (actuated by a passive valve) and long-term decay heat removal is ensured by heat loss from the pressure tubes to the large volume of water available to flood the calandria. The fuel for the dry calandria version is TRISO particles in fuel compacts that are placed in a SiC-coated graphite matrix with coolant channels. The SiC coating is required to protect the graphite from oxidation in high temperature steam. Analyses show that this design is capable of dissipating heat from voided fuel elements without exceeding design limits.

The wet calandria version also has a gas-filled calandria vessel like that in the dry calandria version, but without the flooding capability. The fuel channel for the wet calandria version includes a thin-walled zircaloy tube, which creates an annular space around the calandria tube that is filled with low-pressure, low-temperature light water moderator. This annular moderator acts as a heat sink during both normal operation and during loss-of-coolant events. Heat from the moderator is dissipated passively to the containment atmosphere by natural circulation to reservoirs located on the calandria wall. The fuel for the wet calandria version is a multipin fuel bundle, similar to the CANDU bundle design, but with a SiC-coated graphite plug replacing the center pin and with the traditional Zircaloy fuel cladding replaced by SiC cladding or another corrosion resistant ceramic. The wet calandria version has a relatively flat thermal flux profile, negative coolant and moderator void coefficients and tight neutronic coupling.

2.4.3 High-Conversion Pressure Tube Reactor (W5)

The High Conversion PTR is similar in design to the dry calandria version of the Passive PTR, but there are very limited details on the proposed overall plant design. Like the Passive PTR, the High Conversion PTR has a gas-filled calandria surrounding the horizontal pressure tubes. For this design, flooding of the calandria under accident conditions is achieved passively by gravity feed from a light water reservoir located above the calandria.

The fuel for the high Conversion PTR is a once-through thorium-uranium seed and blanket type fuel. The overall dimensions of the fuel bundles are the same as for normal CANDU fuel; however, to maximize the conversion ratio, the fuel pin diameters are smaller, and the pins are bundled with a tighter pitch. The seed fuel is placed in every fourth pressure tube and consists of 13.5% 235U in a uranium-15%Zr metal matrix. The blanket fuel is BISO-coated thoria (ThO_2) and 5% ^{235}U uranium oxycarbide (UCO) particles embedded in a graphite matrix. Both the seed metal fuel slugs and the blanket-pressed and sintered graphite matrix pellets are clad with Zircaloy. The channels are fueled at a ratio of one seed channel to three blanket channels. The blanket fuel kernels and the seed and blanket enrichments are designed for a blanket fuel residence in the core of 10 years and for leveling of the power density between the seed and blanket channels.

2.5 Supercritical Water-Cooled Reactors

A supercritical light water reactor would operate above the critical temperature and pressure for water (374°C, 221 bar) (705°F, 3208 psia). The key advantages to the concept that are derived from the use of higher temperatures during heat addition include the following:

- Significant increases in thermal efficiency can be achieved relative to current generation LWRs. Estimated efficiencies for supercritical water-cooled reactors are in the range of 40–45%, compared to 32–34% for state-of-the-art LWRs.
- A higher heat transfer rate per unit mass flow results from the large specific heat above the critical point. This leads to (a) a reduction in the reactor coolant pumping power, (b) higher fuel cladding-to-coolant heat transfer coefficients, and (c) reduced frictional losses due to lower steam mass flow rates.
- A lower coolant mass inventory results from the reduced coolant density, as well as a lower reactor coolant system heat content. This results in lower containment loadings during a design-basis loss-of-coolant accident (LOCA)
- No departure from nucleate boiling (DNB or dryout) exists due to lack of a second phase, thereby eliminating heat transfer regime discontinuities within the reactor core. However, an excessive increase in heat flux and/or decrease in coolant flow will cause predictable heat transfer deterioration in supercritical water-cooled reactors.
- Because the coolant does not undergo a change of phase, the need for steam dryers, steam separators, and recirculation pumps, as well as steam generators, is eliminated.
- The high coolant outlet temperatures achievable with supercritical water-cooled reactors may allow these plants to be used to produce hydrogen.

Six supercritical concepts were submitted for consideration, including one concept that has four variants (the supercritical water-cooled CANDU: W6). The concepts are summarized in Table 5 and grouped into four categories: the supercritical, light-water-cooled, thermal spectrum reactor design (W21), the supercritical light-water-cooled, heavy water-moderated reactor designs (W6), supercritical, light-water-cooled fast reactor designs (TWG1), and the marble fuel reactor (W2).

2.5.1 Supercritical Light Water-Cooled Thermal Reactors (W21)

The Japanese supercritical light water thermal spectrum reactor (SCLWR) has been the subject of considerable development work over about the last 10 years. The SCLWR reactor vessel is similar in design to ABWR. High-pressure (250 bar) coolant enters the vessel at 280°C. The inlet flow splits, partly to a down-comer and partly to a plenum at the top of the core to flow down through the core in special water rods to the inlet plenum. This strategy is employed to provide good moderation at the top of the core.

The coolant is heated to 508°C and delivered to a secondary cycle, which looks like a blend of LWR and supercritical fossil technology: high- intermediate- and low-pressure turbines are employed with two re-heaters, as in ABWRs.

Table 5. Proposed Supercritical Water-cooled Reactor Concepts.

Concept/ Organization	Concept Name	Moderator	Rating MWe	Outlet Temp (°C)	Net Efficiency (%)	Comments
W21/ Univ. of Tokyo	Thermal spectrum super-critical water-cooled reactors	H_2O	1700	508	44	Once-through, direct cycle
TWG1	Fast spectrum super-critical water-cooled reactors	H_2O	1500/ Mono-lithic	Varied	38-45	Can burn actinides
W6-1 (Super-critical CANDU)/ AECL	CANDU-X Mark1	D_2O	910	430	41	Indirect cycle, forced circulation
W6-2 (Super-critical CANDU)/ AECL	CANDU-X NC	D_2O	370	400	40	Indirect cycle, natural circulation
W6-3 (Super-critical CANDU)/ AECL	CANDU-ALX1	D_2O	950	450	40.6	Dual-cycle- SCW reactor feeds VHP turbine. VHP turbine exhaust feeds SG with traditional indirect cycle
W6-4 (Super-critical CANDU)/ AECL	CANDU-ALX2	D_2O	1143	625	45	Dual-cycle- SCW reactor feeds VHP turbine. VHP turbine exhaust feeds SG and core inlet regeneration.
W2 (Pebble Fuel)/ PNNL, USA	Pebble bed BWR w/Super-critical Steam	H_2O	200	540	40	Fluidized bed of SiC-PyC-coated UO2 particles in supercritical steam

2.5.2 Supercritical Light Water-Cooled, Heavy Water Moderated Reactors (W6)

The CANDU systems appear to be at a similar level of conceptual maturity as the SCLWR. AECL has investigated both indirect (steam generator) and combined direct cycles using very high-pressure turbines. They have also examined a lower power system with natural circulation on the primary side. These designs are based on many of the standard CANDU features, including horizontal pressure tubes fueled with short fuel bundles and surrounded by a low-temperature heavy water (D_2O) moderator (on-line refueling is possible but not required in these designs). The major innovations in these supercritical CANDU energy systems relevant to current CANDUs are (a) a more compact core design (pressure tube spacing and fuel lattice spacing are adjusted to improve overall cost and safety issues), (b) slightly enriched urania fuel in pressure tube bundles, (c) higher thermal efficiency caused by higher outlet temperatures as well as higher pressures in tubes, and (d) enhanced passive safety systems.

2.5.3 Supercritical Light Water-cooled Fast Reactors (TWG1)

Supercritical water reactors can also be designed to operate as fast reactors. The difference between a thermal and a fast supercritical water-cooled reactor is in the lattice pitch and use of additional moderator material. The fast spectrum reactors use a tight lattice but no additional moderator material, whereas the thermal spectrum reactors need both a loose lattice and additional moderator material in the core. Among fast reactor designs, further distinction is whether the reactor will act as a converter or a breeder.

The Japanese design uses mixed U-Pu oxide fuel consisting of depleted uranium and plutonium discharged from pressurized water reactors. The fuel rods are arranged in a tight triangular pitch without use of ducts around the fuel assemblies. The core arrangement consists of a central inner blanket, inner and outer seeds, a blanket between the seeds, and an outside radial blanket, surrounded by reflector shield assemblies. There is also an axial blanket. This core arrangement was adopted to accommodate use of layers of zirconium-hydride ($ZrH_{1.7}$) between the seeds and blankets. The $ZrH_{1.7}$ layers are clad with stainless steel and are placed in the blanket fuel assembly, one or two fuel rod rows inside from the surface to reduce the power spike in the seed. Calculations show that complete and partial negative void reactivity is achieved using the thin zirconium-hydride layers. Positive reactivity insertion during core flooding is managed by control rods, as in a BWR.

If breeding is not a requirement, a simpler design can be pursued. Other researchers (see Appendix W5) have proposed use of a simple, blanket-free pancake-shaped core with streaming assemblies to make a fuel self-sufficient reactor that retains a hard spectrum to effectively burn plutonium and minor actinides from LWR spent fuel, while efficiently generating electricity. This is a passively safe, high leakage core that can use either fertile or fertile-free fuel, depending on whether the objective is to maximize the actinide burning or maximize plant capacity factors and minimize fuel cycle costs.

2.5.4 Supercritical Light Water-Cooled Pebble Bed Reactor (W2)

This reactor has unique inherent safety features due to the following: (a) ceramic coating layers are used to protect the graphite components in both air and steam at high temperatures (450–1600°C) and (b) the small fuel elements may be able to confine most fission products indefinitely at a temperature of 1600°C, and for several hours at temperatures up to about 2100°C.

Pebble bed reactor fuel elements with an external coating of silicon carbide were tested in a high-pressure water facility (190 bar, 350°C, and PWR water chemistry) for 18 months in Russia. The balls performed well. The uranium loading in a 600-MWt pebble bed reactor core (1-meter radius and 2-meter height) will be about 5.1 metric tons. The fuel pebbles are loaded at the top of the reactor core and are discharged at the bottom. The discharge exposure is about 40,000 MWd/MT. The fuel residence time is about one year. The U^{235} enrichment of the discharged fuel pebbles is about 2.0 wt%.

2.6 High Conversion Water-Cooled Reactors

Most high conversion water-cooled-reactor core concepts are similar in that they use a tight lattice based on a triangular pitch to minimize moderation and produce the fast spectrum essential to achieve a high conversion ratio. Most do this within a BWR, but two designs are based on the PWR. Since the

BWR runs with a void fraction in the core, which can be increased relative to a normal BWR, it can run with reduced moderator density relative to a PWR for the same lattice dimensions. The PWRs must use heavy water, with its decrease in moderating power relative to light water, to compensate and provide a harder spectrum for a given configuration. Other variants are the fuel assembly geometry and the design differences related to concerns over the void coefficient, which tends to be positive in a core with a hard (under-moderated) spectrum. The latter results in most designs using flat cores in order to increase leakage during voiding and thereby make the void coefficient negative. These nuclear energy systems also require recycle of the fissile material.

The features of the various high conversion core designs are summarized in Table 6. Columns 1 and 2 of the table list the acronym used and the principal designer. There are more variations in this concept set, but these represent the ones documented for the Technical Working Group. The third column in the table gives the reactor type, i.e., the nuclear steam supply system used. In general it is the Advanced BWR (ABWR) design that would be used, however, one concept has integrated their core with a more advanced version, ABWR-II, and one intends to use aspects of the Simplified BWR (SBWR) to improve safety. The Safe and Simplified BWR (SSBWR) is an indirect cycle that uses a boiling system and a steam generator to produce steam in the secondary system. It is an integral design and the steam generator is within the reactor vessel. The last two concepts in the table are the integral system PWR (ISPWR) and a loop-type PWR. The ISPWR steam generators are inside the vessel and natural circulation is used.

All the designs listed in the table use tight lattices to harden the spectrum, although the disadvantage is that tight lattices make cooling more difficult. The tight lattices use a triangular pitch in all cases, except for one reduced moderator design (the one using a square fuel assembly), which uses a square pitch. In some designs, as indicated in Column 4, a square fuel assembly is used, and in some it is a hexagonal fuel assembly. The square lattice takes advantage of existing BWR geometry, whereas the hexagonal lattice takes advantage of the more natural geometry using a triangular pitch. Some variants of

Table 6. High-conversion water-cooled-reactor core designs.

Acronym	Principal Designer	Reactor Type	Fuel Assembly (FA) Shape	Coolant[a]	VC Strategy
HCBWR (W9)	Hitachi	ABWR-II	Square	LW	Void tubes
HCBWR-Th (TWG6)	BNL	SBWR/ABWR	Hex	LW	Thorium fuel cycle
SSBWR (W19)	Hitachi	Indirect Cycle BWR; Integral system	Hex	HW changing to LW during the fuel cycle	—
BARS (W27)	Toshiba	ABWR	Square	LW	FA with different heights
RMWR (W24)	JAERI	ABWR	Hex	LW	Double flat core
RMWR (W24)	JAERI	ABWR	Hex	LW	Void tubes
RMWR (W24)	JAERI	ABWR	Square	LW	No blanket
ISPWR (W20)	Mitsubishi	PWR; Integral system	Hex	HW	—
PWR (W30)	Mitsubishi	PWR	Hex	HW	Seed/blanket

[a]LW = light water; HW = heavy water

the standard BWR square fuel assembly have been tried, wherein the external dimensions are increased by a factor of two, similar to the size of a PWR fuel assembly.

In most designs, other features beside the tight lattice are necessary to either reduce moderation further and/or to improve cooling. As mentioned above, the ISPWR and PWR use heavy water to reduce moderation, whereas the BWRs take advantage of the presence of additional void. The SSBWR is the only BWR that also uses heavy water as coolant to loosen the lattice and improve circulation in the core. This is feasible since the SSBWR uses an indirect cycle, and the heavy water remains in a closed loop. This design also uses the spectral shift concept by diluting the heavy water with light water through the fuel cycle in order to lengthen the cycle. The use of heavy water or light water is indicated in Column 5.

Another way to reduce moderation is to use a control rod follower. The water in the gap between the fuel bundles in the top part of the reactor contributes to moderation and the insertion of a follower, which is an inert material, displaces the water without adding absorber. The reactor can also be operated with the follower withdrawn if it is desirable to increase moderation.

One of the problems of designing a core with a fast spectrum is the tendency to have a positive void reactivity coefficient because of the under-moderation. Most designs use a short core (~1 m) to increase leakage and thereby make the void coefficient negative. However, many other design changes have been considered to also increase the negative void coefficient and/or to allow for an increase in core height (and therefore, power). These design features are noted in Column 6 of Table 6.

There are three different reduced moderator water reactor designs with different objectives, and they use different designs to deal with the void coefficient. The different designs are (a) to achieve a high conversion ratio (1.1), (b) to obtain both a high burnup (60 GWd/t) and a 2-year cycle, and (c) to simplify the design. The first design objective is obtained with a double flat core, which consists of a sandwich of two flat cores between three blankets. The second uses void tubes within the core. The third, the square pitch case, uses no blanket.

2.7 Pebble-Fuel Reactors

The light-water-cooled pebble fuel reactor concept can be viewed as a way to combine the attractive characteristics of the high-temperature gas-cooled reactor (e.g., good retention of the fission products at high temperature, passive decay heat removal) with the traditional LWR technology. The pebble fuel reactor concepts submitted to the Generation-IV water-reactor evaluation committee are characterized by use of spherical fuel particles (outside diameter in the 1 to 10-mm range) with either a ceramic or metallic cladding. The particles are kept in suspension in the core by the water coolant flow as a fluidized bed. If a loss-of-flow or loss-of-coolant accident occurs, the fuel particles fall into a subcritical configuration that automatically shuts down the reactor. Moreover, because of the large surface-to-volume ratio, the fuel normally operates at relatively low temperatures.

A summary of the general characteristics of the three pebble fuel reactor concepts submitted to the Generation-IV water- reactor evaluation committee is reported in Table 7.

Two subgroups can be identified within the pebble fuel reactor class:

- Concepts with TRISO particle fuel
- Concepts with zirconium-clad fuel.

A brief description of these two subgroups is presented below, respectively.

Table 7. Summary of the pebble bed concepts submitted to DOE for the Generation-IV Program.

Gen-IV Designation	Proponent	Size	Coolant State	Mode of Circulation	Cladding
W1	Tsiklauri (PNNL, USA)	200 MWe	Boiling (7.0 MPa)	Direct	TRISO
W2	Tsiklauri (PNNL, USA)	240 MWe	Supercritical (24 MPa)	Direct	TRISO
W4	Sefidvash (UFRGS, Brazil)	1 MWe per assembly	Pressurized (15 MPa)	Indirect	Metallic Zr

2.7.1 Pebble-Fuel Reactors with TRISO Fuel (W1, W2)

These are direct-cycle reactors with a fludized-bed core made of several million TRISO coated fuel particles. The fuel elements are small pebbles (between 2 and 10 mm diameter) consisting of low-enrichment UO_2 or UCO kernels coated with 3 layers. The inner layer is made of porous pyrolytic carbon (PyC) called the buffer layer, providing room for fuel swelling and gaseous fission product accumulation. The second layer is a dense PyC coating; the outer layer is a corrosion resistant silicon carbon coating (SiC).

For concept W1, boiling water is both the coolant and the main moderator in this reactor, although the carbon in the PyC and SiC provides some moderation as well. The fuel elements, containing 4.8% enriched uranium, are loaded at the top of the reactor core and are discharged at the bottom, without need for shutdown and depressurization.

This reactor has very strong negative coolant temperature and void coefficients of reactivity. The fuel temperature reactivity coefficient is also strongly negative. Core reactivity is managed by means of movable gas-cooled control rods inserted from the core bottom. About 140–150 control rods with a spacing of about 12 cm are required for the reactor.

This core is designed as a frustum cone with the bottom being a perforated coolant dispenser and the upper cap being a perforated plate that constrains the fuel particles. Therefore, the fuel is contained between the outer conical case and the perforated bottom and upper plates. The coolant flow path is as follows. Water coolant from jet pump nozzles enters the lower plenum, flows through the perforated coolant dispenser into the pebble bed. The water cools the pebble bed as it is heated and boils, while moving upward. The two-phase mixture exits the core through the perforations in the upper plate and enters the outlet plenum, located above the core. The cross section of the frustum cone increases vertically to compensate for void fraction increases and keeps the coolant velocity low. The balance of plant of the reactor is similar to standard BWR designs.

The capability of the TRISO fuel particle to retain the fission products at high temperature enhances the performance of the Pebble Bed BWR under severe accident conditions. Also, in case of complete loss of coolant the decay heat could be conducted radially across the core. Note that the fission products silver and palladium diffuse through pyrolytic and silicon carbide coatings. In the gas reactors operated to date, those fission products generally remained in the graphite matrix of the compacts. In this concept, they may be released to the coolant.

Concept W2 is virtually identical to the Pebble-Bed BWR (Concept W1) except that the water coolant operates at supercritical pressures and temperatures. (Supercritical water-cooled reactors are discussed above, and this concept is included in that concept set as well.) This eliminates the phase change within the core and the need for steam separators and dryers, as well recirculation and jet pumps. Also, higher thermal efficiencies (up to 45%) can be obtained with this approach.

2.7.2 Pebble-Fuel Reactors with Zircaloy-Clad Fuel (W4)

The reactor core is made of a variable number of modules, each generating about 1 MWe. Each basic module has a core region and a steam generator in its upper part, and a fuel chamber and pump in its lower part. The core region consists of a 25-cm-diameter fluidizing tube in which, during reactor operation, the spherical fuel elements are kept in suspension by the upward coolant flow. The fuel chamber is a 10-cm-diameter tube, which is directly connected underneath the fluidizing tube. A neutron absorber shell slides inside the fluidizing tube, acting similarly to a control rod, for the purposes of long-term reactivity control.

The operating pressure and temperature are the same as a traditional PWR. However, a steam generator of the shell-and-tube type is integrated into the upper part of each module. The pump circulates the water coolant inside the module moving upward through the fuel chamber, the fuel region, and the steam generator. Then the coolant flows back down to the pump through the concentric annular passage. Each module is provided with a pressurizer to keep the pressure constant.

The 8-mm diameter spherical fuel elements are made of slightly enriched uranium dioxide, clad with Zircaloy. The coolant velocity is selected to fluidize the particles so that the core operates at the reactivity maximum in the reactivity vs. moderator-to-fuel-ratio curve. That is, any deviation from the reference coolant flow level results in a reactivity decrease that automatically shuts down the reactor. In case of a complete loss of flow or coolant, the fuel particles fall down into the fuel chamber, which is a sub-critical configuration. Then the fuel chamber is cooled by natural convection transferring heat to the surrounding air or water pool.

2.8 Advanced Light Water Reactors with Thorium/Uranium Fuel

Five general approaches for fueling an advanced water-cooled-reactor nuclear energy system with thorium are considered: (1) the once-through seed and blanket (Radkowsky) thorium fuel design, (2) the high conversion light water reactor with seed and blanket thorium fuel and U-233 recycle, (3) once-through homogeneous thoria-urania (ThO_2-UO_2) fuel, (4) once-through micro-heterogeneous thoria-urania fuel, and (5) metal-matrix thoria-urania dispersion fuel. Each of these concepts is described in the following subsections.

2.8.1 Advanced Light Water Reactors with Once-Through Seed and Blanket Thorium Fuel

There are a number of ways thorium can be used in current and future LWRs. Probably the best-known once-through thorium fuel-cycle concept was developed by Dr. Alvin Radkowsky and associates in Israel and is known as the Radkowsky Thorium Fuel Cycle. The concept is based in part on the ideas and experiences of the Bettis Atomic Power Laboratory's Light Water Breeder Reactor (LWBR) program as implemented and successfully demonstrated at the Shippingport reactor in the 1980s. However, in contrast to the LWBR, the Radkowsky concept assumes a once-through thorium fuel cycle with no recycling; the U-233 that is bred is mostly burned in situ, and the fuel rods that contain the U-233 (which is denatured by nonfissile uranium isotopes) are then disposed of.

The main idea of the Radkowsky thorium fuel cycle is utilization of a seed-blanket unit (SBU) that is fully interchangeable with current LWR fuel bundles. The SBU geometry allows a spatial separation of the uranium (mostly in the seed) and thorium (blanket) parts of the fuel bundle. The central region of the assembly (seed) includes uranium enriched to a maximum of 20%, while the external region of the assembly (blanket) includes natural thoria (ThO_2) spiked by a small amount of 20% enriched urania (UO_2). This arrangement provides the necessary flexibility for designing the seed as an efficient supplier of well-thermalized neutrons to a subcritical blanket that, in turn, is designed for efficient generation and in situ burning of U-233. This approach has been applied to both VVER and PWR core designs with considerable success and could also be applied to other water reactor designs in the future (e.g., BWRs or small modular light water reactors currently under development). One variant of this approach uses plutonium rather than uranium as fuel. This improves the nonproliferation characteristics of the concept by virtue of being able to dispose of large amounts of plutonium.

2.8.2 High Conversion Light Water Reactors with Seed and Blanket Thorium Fuel and U-233 Recycle

LWRs attained economic significance during the mid-1960s for central power station electricity generation on the basis of relatively low capital and uranium costs, abundant enrichment capacity, and strong technical support from the U.S. Naval Reactor Program. However, the subsequent development sequence of nuclear power in the world was not what had been originally envisioned. Originally, it was expected that a modest number of LWR plants would be built, providing needed power, the technical basis for a growing nuclear industry, and the fuel for fast spectrum breeder reactors. The fast spectrum breeder reactor was expected to provide the basis for a fuel self-sufficient (plutonium recycle based) nuclear power industry. However, the commercial breeder reactor was not fully developed, and the LWR was a much stronger commercial competitor for power plant construction versus fossil fuels (in the 1970s and early 1980s) than originally expected. The result is that, worldwide, we have a large number of LWRs without a long-term sustainable fuel cycle. The current once-through uranium fuel cycle is essentially transitory, i.e., it had a beginning and it will end not too far in the future.

However, it is possible to design and build a thermal spectrum LWR with a fully self-sufficient fuel cycle if the U-233/Th-232 fuel cycle is adopted. The primary advantage of using U-233 fissile material in thermal reactors is that the average number of neutrons produced per atom of fissile material destroyed is large enough for fuel self-sufficiency, whereas, if either U-235 or Pu-239 is used in a thermal spectrum reactor, the average number of neutrons produced per atom of fissile material destroyed is too small for fuel self-sufficiency. The Th-232 is needed to produce the U-233, of course. The light water breeder reactor (LWBR) can be similar to current pressurized water reactors (PWRs) and can take advantage of all the technology that has been developed to support the PWR. However, its core design must be slightly different so as to better conserve neutrons. Specifically, separate seed and blanket fuel regions are used to maximize the neutron production, the reactor is controlled by moving the seed (with PWR type control rod drives) rather than inserting absorber rods so as to eliminate parasitic neutron losses, blankets and reflectors are located to minimize leakage, and the fuel rods are spaced relatively closely.

Thorium, which averages 7.2 parts per million in the earth's crust, is the 39th most abundant of the 78 crustal elements. It is about three times more abundant than uranium. When bred to the fissile U-233, thorium releases about the same energy per unit mass (79 TJ_{th}/kg) as uranium bred to Pu-239 (80.4 TJ_{th}/kg). Thorium and its compounds have been produced primarily from monazite, where it is produced as a by-product of the recovery of titanium, zirconium, tin, and rare earths. Only a small portion of the thorium produced has been consumed. Limited demand for thorium, relative to the demand for rare earths, has continued to create a worldwide oversupply of thorium compounds and mining residues. Thus, in the short term, thorium is available for the cost of extraction from rare-earth processing wastes. In the longer term, large quantities of thorium are available in known monazite deposits in India, Brazil, China, Malaysia, and Sri Lanka.

The existing LWRs convert some fertile U-238 or Th-232 into fissile fuel; however, the overall nuclear resource utilization is only about 1% of the energy potentially available from the mined ore. Based on the use of a well-established and successful LWR technology and the potential for an assured energy supply for a very long time, development of the LWBR U-233/Th-232 fuel cycle appears to be an attainable and important alternative for future energy generation.

2.8.3 Advanced Light Water Reactors with Once-Through Homogenous and Micro-Heterogeneous Thoria-Urania Fuel

A third approach for using thorium in current and future LWRs is use of high burnup homogenously mixed thorium-uranium dioxide (ThO_2-UO_2) fuels. In this case, the thoria and urania are mixed uniformly, and the fuel rods and bundles have essentially the same geometry as current LWR fuel. Fuel with 75% thoria and 25% urania (enriched with U-235 to slightly less than 20%) can reach burnups of about 54 MWd/kg initial-heavy-metal. Fuel with 65% thoria and 35% urania can reach burnups of about 75 MWd/kg. A variation on this approach was developed during the LWBR program and more recently investigated at MIT and includes some small amount of what is called *micro-heterogeneity*. Here, the fuel form might be a duplex pellet with the urania on the inside and the thoria on the outside, or it might be a fuel rod with alternating short stacks of thoria and urania pellets, or it might be alternating thoria and urania fuel rods. Providing some small separation between the uranium and thorium improves the core reactivity and achievable burnup.

2.8.4 Metal-Matrix Thoria-Urania Dispersion Fuel

Metal-matrix thoria-urania dispersion nuclear fuels are composed of a fine dispersion of thoria-urania micro-spheres in a zirconium metal matrix. About 50% of the oxide is thoria, and about 50% is urania. The oxide fuel to metal matrix ratio is also about 1 to 1. The uranium enrichment is about 19.5%. The pure zirconium matrix provides fuel and fission product containment, high thermal conductivity, and superior corrosion resistance during long reactor service and also during waste storage. The thermal conductivity of the metal matrix greatly enhances heat removal. This can allow higher fuel ratings and fuel surface temperatures for use in supercritical water-cooled reactors and other advanced Generation IV reactors.

The potential benefits that may be gained with this proposed fuel form include low fuel fabrication costs due to the production of long-length rods by a metal drawing process, high actinide burnup, inherent proliferation resistance, improved irradiation stability due to low internal fuel temperatures and stored energy, and high waste stability. The potential for high actinide burnup exists because the buildup of the U–233 during irradiation of the Th-232 can significantly extend the fuel residence time. Also, as a once-through system, this fuel is designed to be disposed after irradiation without processing and without encapsulation. The zirconium alloy matrix, Zircaloy shell, and Zircaloy cladding combine to form an excellent waste containment system.

2.9 Advanced Water-Cooled-Reactors with Dry Recycling of Spent LWR Fuel

The proliferation-resistant, dry recycle of spent light water reactor (LWR) fuel into either heavy water reactors (HWRs) or LWRs addresses many of the Generation IV objectives. The technology consists of: (a) chopping spent LWR fuel into small segments, (b) exposing the fuel to successive oxidation and reduction heating cycles, (c) refabricating LWR or CANDU type fuel using the powder produced by the oxidation/reduction cycles, and (d) reirradiating. Application of this technology to either

LWR/HWR or to LWR/LWR recycle has many similarities, although there are some important differences. The residual fissile material in typical LWR spent fuel is sufficient to power a CANDU reactor; however, when the dry recycle material is used again in a LWR, fissile material must be added. The term *DUPIC*, for “dry use of spent PWR fuel into CANDU” has been coined for the application of this technology for light-to-heavy water reactor recycle; the term *AIROX*, for “Atomics International Reduction Oxidation” has been used for the process as applied to LWR/LWR recycle.

The technology may be particularly important and effective in addressing the accumulation of spent fuel in many countries, and, in particular, in the United Sates. Delays in developing geological repositories and hurdles in obtaining licenses for either new spent fuel storage facilities or for expanding existing facilities threaten to limit the extent to which new reactors of any design can be introduced. Furthermore, siting of a second repository, or a significant expansion of the capacity of the planned repository, would be required in any growth scenario for nuclear power in the United States, and that would be a formidable challenge.

Other benefits of dry recycle are its high degree of proliferation resistance; it is expected to be cheaper than conventional recycling and MOX fuel fabrication and to be cost effective compared to direct disposal; and it can effectively utilize ex-weapons fissile material (either plutonium or high enriched uranium). In the case of LWR/HWR recycle (DUPIC), it would significantly reduce uranium requirements compared to the once-through HWR fuel cycle, and would reduce the heat load and cost of spent fuel disposal in a geological repository.

2.10 Advanced Light Water Reactors with Plutonium and Minor Actinide Multirecycling

These are ALWRs with either normal moderation or reduced moderation cores that burn plutonium and minor actinides. Plutonium and minor actinide multirecycling in Generation 4 water-cooled reactor nuclear energy systems has the potential to reduce high-level waste burdens, extend uranium resources, reduce enrichment requirements, and therefore improve the sustainability of nuclear power. Use of plutonium in LWR cores requires careful attention to the issues of maintaining criticality to high burnup, neutron energy spectrum hardening, control rod effectiveness, core transients, void reactivity coefficient, power peaking, and safeguards against diversion of fissile materials. Minor actinide recycling would be most effective, with an improvement of the decontamination factor achieved during reprocessing to minimize the fraction of minor actinides that escape the cycle and go to waste disposal. Effective shielding or remote handling will be required for a minor actinide recycle fuel fabrication facility.

A number of fuel designs have been developed for plutonium and minor actinide multirecycle, some of which are MIX, CORAIL, APA, and RMWR. The MIX concept uses a homogeneous mixture of oxides (UO_2 and PuO_2) in each fuel rod. The CORAIL concept uses a heterogeneous arrangement of UO_2 rods and MOX rods, and the APA concept uses a heterogeneous arrangement of UO_2 rods and rods with PuO_2 in an inert matrix. The Reduced Moderation Water Reactor (RMWR) uses a special core shape and tight lattice to attain high conversion ratios (CR ~ 1), while maintaining a negative void reactivity coefficient.

Except for MIX, these core designs are mainly at early stages. Much additional R&D is needed on the details of the fuel assembly design, safety analyses, reprocessing, fuel fabrication, and cost estimates.

3. CONCEPT EVALUATION

The Technical Working Group conducted a comprehensive evaluation of the 38 water-cooled reactor concepts identified in order to determine their potential to achieve the Generation IV goals. This evaluation was used as a foundation for the initial screening step (Screening for Potential) in which any candidates determined to have insufficient potential for subsequent Technical Working Group recommendation for Generation IV Program R&D support were eliminated from further consideration. The evaluation process and conclusions are described in the following sections.

3.1 Evaluation Process

The technical working group evaluation process consisted of the following main activities, conducted by Technical Working Group members individually, in subgroups, and in several full Technical Working Group meetings over the period of April through July 2001:

- *Identification and Information Gathering.* Most of the water-cooled reactor concepts were identified through the DOE Request for Information process. Submittals to DOE included, in most cases, a substantial technical information about the concepts. In some cases, additional information was requested and obtained from the respondents. As discussed above, this information was compiled into a master table to facilitate broad understanding and initial organization of the information.[b]

- *Organization into Concept Sets.* The concepts were consolidated into ten sets, based on common key features or design characteristics. (The sets are described in the previous section and in the appendices to this report.) This step was very important in making possible a relatively efficient evaluation process.

- *Subgroup evaluation.* Each set was assigned to a subgroup of the technical working group, consisting of two or three technical experts. The subteam examined the information available, obtained additional information as needed, and assembled a preliminary assessment. Each subgroup then briefed the full technical working group on the concept set and presented its preliminary findings at the June 2001 Technical Working Group meeting.

- *Full team evaluation.* Over the course of a 3-day technical working group meeting held during August 2001, the entire Technical Working Group evaluated and, in large measure, reached consensus on the evaluation for each concept set. The full Technical Working Group evaluations explicitly addressed:

  - Assessment potential, with respect to each Generation IV goal and sub-goal

  - Assessment of individual reactor concepts within the set, in areas where there were significant differences from the overall set.

  - Identification of research needs.

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b. The master table summarized the following information on each concept: reactor size, plant design approach, coolant and moderator and their physical state, cycle, thermal efficiency, reactivity control, primary system layout and mode of circulation, neutron spectrum, fuel form, cladding materials, irradiation cycle and refueling, decay heat removal system, containment, important safety characteristics, proliferation characteristics, resource utilization, economic characteristics, and R&D needs. The Master Table is provided in Appendix W12 of this report.

- Scoring, by producing a scorecard showing in graphical form the potential of the set against each of the Generation IV goals.

- *Documentation.* The Technical Working Group evaluation for each concept set was documented, in detail, through compiling the concept descriptive material and summaries of each set evaluation, including R&D requirements, and tabulating major advantages and disadvantages, factors influencing scoring, and completed scorecards. Each of these compilations was produced as a stand-alone document; each is included as an appendix to this report.
- *Screening for Potential.* This final screening was accomplished, via extensive team discussion and consideration of previous evaluation work, during the August technical working group meeting.

3.2 Evaluation Results for Each Concept Set

Following are capsule summaries of the evaluations for each concept set.

Integral Primary System Reactors

These light water reactor concepts are characterized by a primary system that is fully integrated in a single vessel, which makes the nuclear island more compact and eliminates the possibility of large releases of primary coolant. The emphasis is on utilization of existing LWR technology, plant simplification, modularity, elimination of accident initiators, and passive systems to cope with the consequences of accident events. Of the three Generation-IV high-level goals, this class of reactors mainly addresses the potential for superior safety and good economics. However, the sustainability of the IPSRs can also be better than the reference due to the flexibility in the fuel cycles, featuring options of straight burn, higher burnup, and MOX fuel. The sustainability is also better than the reference due to the flexibility in the reactor core and fuel designs, featuring options of a once-through low-enriched uranium fuel or longer fuel cycles and higher burnup or MOX fuel. At this point, the key R&D issues for these systems appear to be the economic viability of a modular design approach, as well as the reliability and design of the in-vessel components.

Advanced Loop Pressurized Water Reactors

The common innovative characteristic of these reactor designs is the use of a safeguard vessel (or series of vessels and pipes) that envelops the whole primary system for mitigation of primary system component failure. Moreover, adoption of the additional vessel enables elimination of some safety systems. This reactor concept offers potential for superior safety compared with the reference LWRs. However, issues to be resolved include reliability and maintenance of the primary system components, which are not easily accessible, and the impact of the additional vessel on capital cost.

Simplified Boiling Water Reactors

These are various size boiling water reactors with natural circulation in the core region, no recirculation pumps, and, in most cases, highly passive decay heat removal systems. With one exception (the SMART concept), the concepts within this group are all founded on existing and proven BWR technology and do not need extensive R&D for their deployment. They feature various design improvements intended to provide economic or other advantages. At this point, the key R&D issue for these systems appears to be to demonstrate their economic value relative to other designs.

Pressure-Tube Reactors

This concept set is based on the CANDU design. Emphasis is on improving the economics of current generation CANDU reactors by replacing the heavy water coolant with light water, moderately increasing the thermal efficiency, and simplifying and reducing the size of the nuclear island. Some concepts also adopt thorium fuel, which would result in better proliferation resistance and somewhat better natural resource utilization. The key R&D issues associated with development and commercialization of these systems appear to be the feasibility of pressure tubes operating at higher temperatures and pressures, and, for those designs with thorium fuel, the reliability and economic performance of the fuel.

Supercritical-Water-Cooled Reactors

The unique thermo-physical properties of supercritical water offer potential for designing nuclear reactors with significantly higher thermal efficiencies and considerable plant simplification, compared to the ALWR. However, to make such systems technologically feasible, advances are required in high-temperature materials to improve corrosion, stress corrosion cracking, and wear resistance, in neutronics to improve fuel-cycle versatility with these advanced materials, and in neutronics and thermal-hydraulics to ensure an acceptable level of safety and stability.

High-Conversion Water-Cooled Reactors

The high conversion cores have the advantage of greatly increasing fuel utilization. Since uranium resources will eventually become scarce, this concept set has the potential of being very important in the future. However, there are key R&D issues to be addressed, including the safety of the core (e.g., negative void reactivity coefficient), development of appropriate fuel cladding and core internals structural materials, demonstration of effective coolability of tight cores, and development of suitable proliferation resistant fuel recycling techniques to take advantage of the increased production of fissile material.

Pebble-Fuel Reactors

Emphasis in this class of reactors is on passive safety (i.e., all concepts feature passive shutdown and also passive decay heat removal capabilities) and reduced fuel temperature operation because of the large heat transfer area available for removing the nuclear heat. However, the fuel proposed for these concepts has not been tested to any great extent. Although the committee decided to retain this concept set for further evaluation, some of the committee believe that it is one of the less promising candidates in the Generation IV water-reactor group.

Advanced Light Water Reactors with Thorium/Uranium Fuels

The significant advantages of the once-through thorium cycles with respect to proliferation resistance and waste form stability are very attractive to society as a whole but provide little incentive to the current nuclear fuel industry. The energy resource sufficiency advantage of the U-233/Th-232 light water breeder reactor fuel cycle is currently out weighted by reliability and cost issues. However, farther out in the future our low-cost uranium supplies will become depleted and the thorium fuel cycles will eventually become cost effective.

Advanced Water-Cooled-Reactors with Dry Recycling of Spent LWR Fuel

These technologies have significant potential for reducing spent fuel volumes, increasing fuel utilization, reducing proliferation risk in recycle, and in enhancing long-term sustainability. Furthermore, they can be employed in both existing reactors and in next-generation reactors, complementing the benefits from those reactor designs. Key R&D issues identified for this fuel cycle include development of cost-effective fabrication processes and equipment and development of adequate solutions for capture and immobilization of the volatile fission products released during the recycling process.

Advanced Light Water Reactors with Plutonium and Minor Actinide Multirecycling

Multirecycling of the plutonium and minor actinides has potential to reduce the high-level waste burdens, extend uranium resources, reduce enrichment requirements, and, therefore, improve the sustainability of nuclear power. However, use of plutonium in LWR cores requires careful attention to the issues of maintaining criticality to high burnup, neutron energy spectrum hardening, control rod effectiveness, core transients, void reactivity coefficient, power peaking, and safeguards against diversion of fissile materials. Effective shielding or remote handling will be required for a minor actinide recycle fuel fabrication facility.

3.3 Research and Development Needs

In the course of the evaluation of the eight concept sets outlined above, a preliminary set of R&D needs was compiled. In all cases, the identified needs were common to two or more concept sets. Table 8 summarizes the primary needs (in preliminary form) versus their corresponding concept sets:

Table 8. Preliminary R&D needs.

Concept Set	W1	W2	W3	W4	W5	W6	W7	W8	W9	W10
R&D Need										
Fuel cladding materials development	X	X	X		X	X	X	X	X	X
Fuel development		X		X			X	X	X	X
Corrosion/fission product transport		X			X					
Structural/core materials development	X	X		X	X					
Optimized core neutronics designs, fuel cycle analysis			X	X	X	X		X	X	X
Core thermal/hydraulic analysis					X	X	X	X		X
PRA – safety and accident analysis	X	X	X	X	X	X	X	X	X	X
Economic analysis	X		X	X		X		X		X
Development of reliable/low maintenance components for long fuel cycles and difficult access	X	X								
Development and validation of neutronic, thermal-hydraulic, and fuel behavior models and data					X	X	X	X	X	X
Confirmation testing of performance and safety parameters	X	X	X	X	X	X	X			

3.4 Evaluation Comparisons

Although the primary purpose of the concept set evaluations at this stage of the Generation IV process is to perform an initial assessment of set potential and to support an initial screening for potential, it is possible to compile from the individual set evaluations some initial sense of their relative merits as well. The following charts shown in Figures 1, 2, and 3 illustrate the scoring of each of the nine sets against each of the eight major Generation IV goals. Use of an “E” means that the concept set is essentially equivalent to the reference ALWR.

3.5 Screening for Potential

One concept was discarded. Concept W15, the U-Np-Pu cycle, was deemed not feasible for large-scale production of electricity because of the scarcity of neptunium and because of the high value of neptunium for alternative uses (e.g., target material for production of Pu-238 for space exploration). A discussion of the rationale for this decision can be found in Appendix W11.

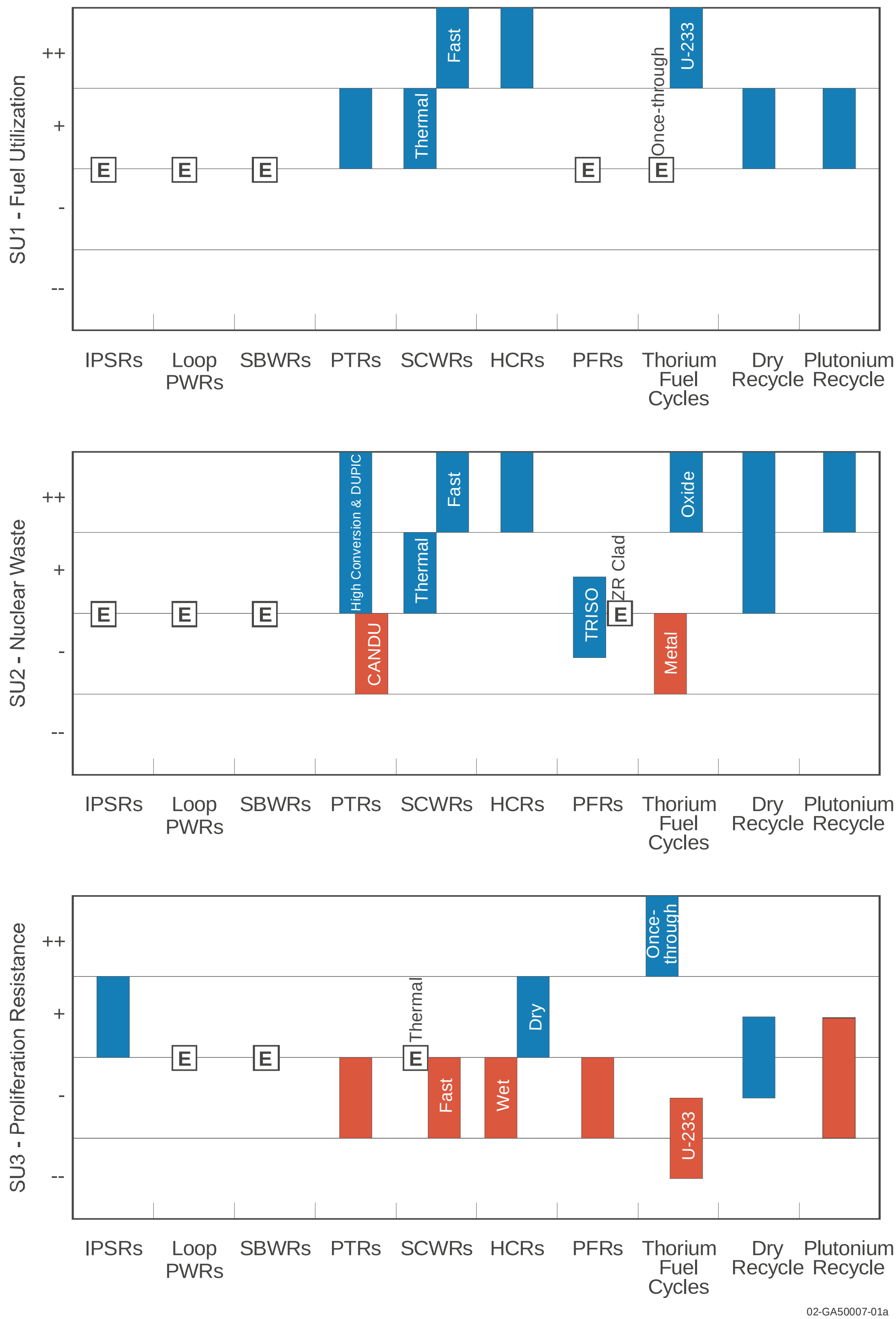

Figure 1. Initial evaluation of performance versus the sustainability goals (fuel utilization, nuclear waste minimization, and weapons material proliferation resistance) for the nine concept sets.

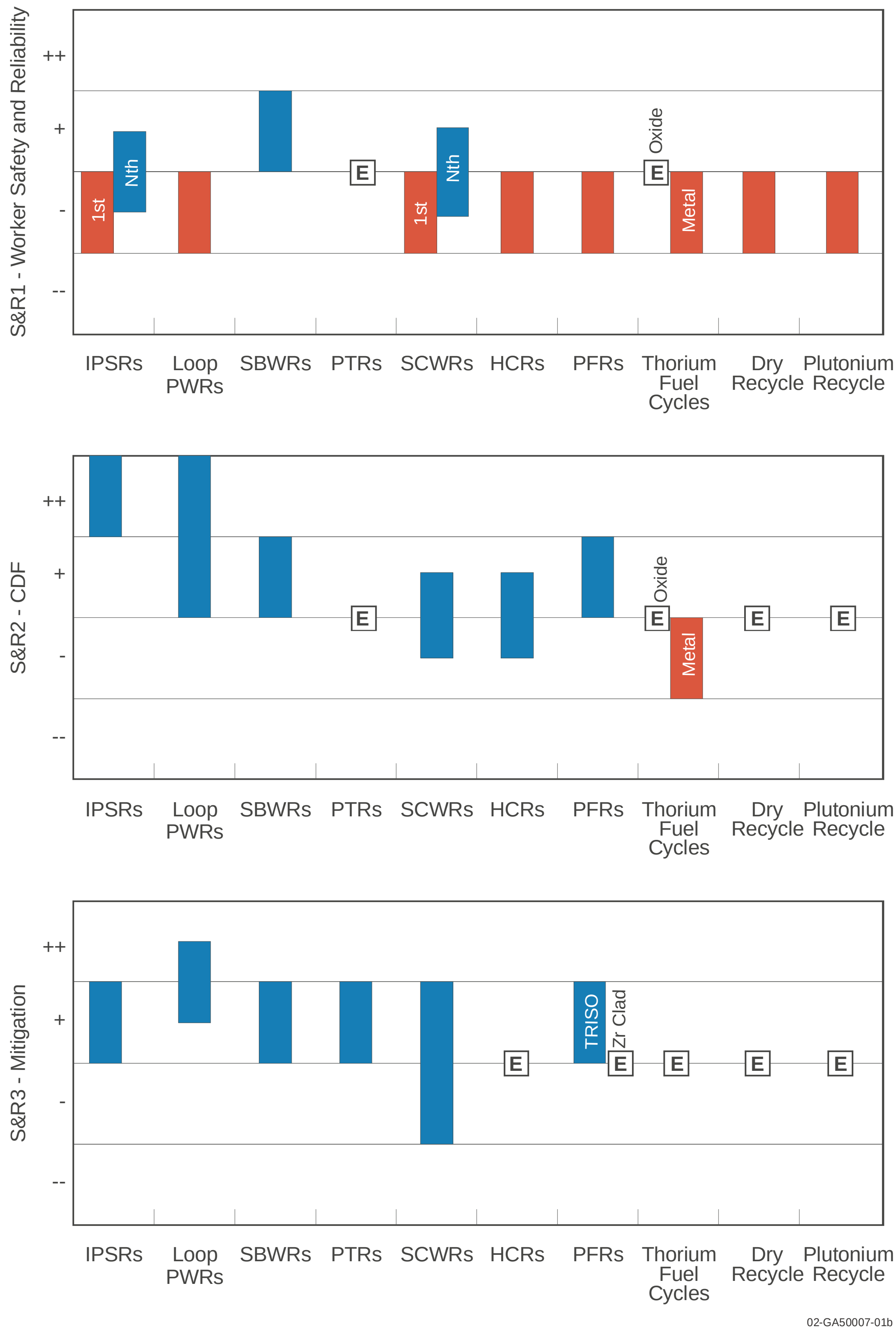

Figure 2. Initial evaluation of performance versus the safety and reliability goals (worker safety and plant reliability, core damage frequency, and accident consequences) for the nine concept sets.

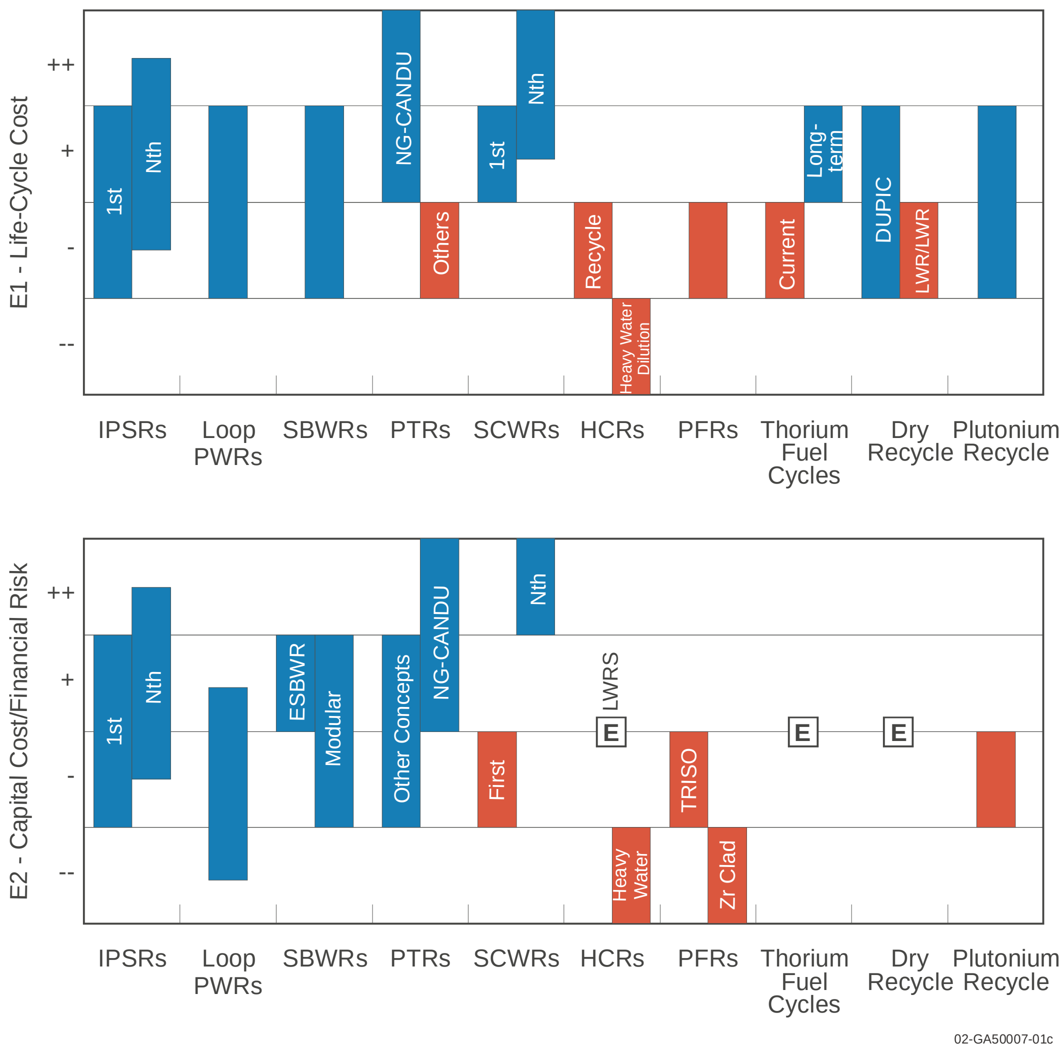

Figure 3. Initial evaluation of performance versus the economic goals (life-cycle costs and plant capital costs and risk) for the nine concept sets.

4. CONCLUSIONS

The 38 reactor and fuel cycle concept submittals in the area of water-cooled nuclear systems were grouped into ten concept sets. The goal was to identify essential discriminating characteristics and potentials, to group and rationalize the concepts, and to identify R&D needs for the concept sets. The objective was not to promote or reject any specific individual or corporate idea or product.

The ten concept sets are the integral primary system reactors, advanced loop-type pressurized water reactors, simplified boiling water reactors, pressure-tube reactors, supercritical water-cooled reactors, high-conversion water-cooled reactors, pebble fuel reactors, ALWRs with thorium/uranium fuels, advanced water-cooled-reactors with dry recycling of spent LWR fuel, and ALWRs with plutonium and minor actinide multirecycling.

All ten concept sets were retained for further assessment in the second phase of the Generation IV Roadmap, while one concept was discarded. Concept W15, the U-Np-Pu cycle was deemed unfeasible for large-scale production of electricity because of the scarcity of neptunium supplies and because of the high value of neptunium for alternative uses (e.g., target material for production of Pu-238 for space exploration).

From a technical evaluation of these concept sets, several R&D needs were identified, which include development of fuel cladding and structural materials for higher burnup and temperature applications, development of reactor components for infrequent maintenance, quantitative assessment of the benefits of small-power modular designs, experimental verification of the passive safety system performance, and updating/validation of existing predictive tools for the expanded design envelope of the advanced reactors.

The next step for the Advanced-Water-Cooled-Reactor Systems Technical Working Group is to perform a quantitative assessment of each of the concepts and/or concept sets, identify the most promising concepts or concept sets and prioritize them, then clearly define the scope of the R&D needed to support deployment of the promising concepts or concept sets.

Appendix W1

Integral Primary System Reactors Concept Set Report

December 2002

CONTENTS

FIGURES

TABLES

ABSTRACT

Seven reactor concepts were submitted to the DOE-NE Request for Information that are characterized by a primary system that is fully integrated in a single vessel, which makes the nuclear island more compact and eliminates the possibility of large releases of primary coolant during a pipe break. These Integral Primary-System Reactors (IPSRs) are based on an indirect-cycle heat-transport scheme. The coolant/moderator is light water, either pressurized or boiling. The primary-coolant mode of circulation is either forced or natural. All the proposed concepts are thermal reactors and make use of low-enrichment-uranium oxide-fuel, clad with zircaloy. Some of the concepts are also designed for plutonium-uranium oxide (MOX) fuels and/or thorium-based fuels.

The emphasis in this class of reactors is on utilization of existing light water reactor (LWR) technology, plant simplification, modularity (e.g., standardization, transportability), elimination of accident initiators, and passive systems to cope with the consequences of accident events. Even though long life cores are not an intrinsic feature of the IPSR, many of the concepts adopt long irradiation cycles for an increased plant capacity, reduced fuel handling, and improved proliferation resistance.

Of the three Generation-IV high-level goals, this class of reactors mainly addresses the potential for superior safety and economics. On the other hand, resource utilization and proliferation resistance are rated as comparable (or just slightly better) than current LWRs with similar fuel cycles.

W1.1 INTRODUCTION

W1.1-a. Background and Motivation for the Concept

In the past decade, light water reactor (LWR) technology has achieved full maturity. LWR plants in the United States and abroad routinely display outstanding performance in terms of safety, capacity factors, and operating costs, which, combined with the current energy crisis in North America and elsewhere, has generated a worldwide resurgence of interest for nuclear power. This accumulation of experience and success constitutes an asset for those Generation-IV reactor concepts that are based on LWR technology.

The Integral Primary-System Reactor (IPSR) concepts seek to maximize the use of existing LWR technology, which is engineered in innovative ways to improve safety and simplify the plant. The main characteristic of these reactors is the integration of the whole primary system within a single pressure vessel. Because a catastrophic failure of the vessel is considered to be incredible, this eliminates (by design) the most important postulated accident for current LWRs, the large release of primary coolant from the rupture of an external-loop pipe (a large loss-of-coolant accident or LOCA). More generally, these reactors are characterized by the adoption of the so-called “safety by design” approach, i.e., an attempt is made to eliminate or reduce the possibility of the main accident initiators by design rather than having to mitigate the consequences of those accidents. For example, integration of the primary-system makes it easier to achieve a higher degree of natural circulation of the primary coolant, which makes loss-of-flow accidents benign. Similarly, the utilization of in-vessel control-rod drives eliminates the possibility of control-rod ejection accidents.

These are small modular reactors, generally with a power of 150 MWe or less; even for the largest plant of the set, the upper power limit is about 1000 MWt (~350 MWe). Their cost basis may, therefore, be different than the current large monolithic plants, and an economy of multiple factory built modules is claimed to take the place of the economy of scale usually associated with big monolithic plants. It must also be noted that there may be conditions (e.g., developing countries with a limited grid, or a developed country where only a small additional increment of capacity is needed) where a 350 MWe or less plant size is more appropriate than a large plant.

W1.1-b. National and International Interest

Over the past several years IPSRs have gained considerable interest within the US and internationally, as testified by the number and origin of proposed concepts that fall into this class (see Table 1), i.e., a total of seven reactors, 3 from the United States, 2 from Japan, 1 from Korea and 1 from Argentina. The best known of the most recent concepts under development is probably the IRIS reactor, initiated by the Westinghouse Electric Co., Massachusetts Institute of Technology, and University of California at Berkeley through a DOE Nuclear Energy Research Initiative (NERI) grant and currently being developed by a collaboration of about 18 research and industrial partners in nine countries. IRIS is one of the four reactor concepts currently being evaluated by the NRC for early deployment in the United States. CAREM, a project of the Argentina’s Commission Nacional de Energía Atómica (CNEA) was initiated over 15 years ago and was used as a reference design in a recent joint-study performed by the International Atomic Energy Agency of the United Nations (IAEA), OECD-NEA and OECD-IEA. Several experimental facilities have been constructed to test various aspects of the CAREM concept. The SMART design being developed in Korea is also widely known and has been the subject of various international studies. Korea has recently announced that a prototype of the SMART reactor will be built starting in 2002. The Multi-Application Small Light Water Reactor (MASLWR) is also being studied

Table 1. Summary of integral primary-system concepts submitted to DOE for the Generation-IV Program.

Gen-IV Designation	Proposer	Size	Coolant State/Pressure	Mode of Circulation*	Containment	References
W10 (SMART)	Chang (KAERI, South Korea)	330 MWth	Pressurized, 15.0MPa	Forced	Spherical guard vessel with suppression pool plus traditional containment	Chang et al. 1999 and 2001; Bae et al. 2001
W14 (CAREM)	Beatriz-Ramilo (CNEA, Argentina)	100-150MWe	Pressurized, 13.0 MPa	Natural	With suppression pool	Delmastro et al. 2000 and 2001; Mazzi et al. 2001
W16 (PSRD)	Ishida (JAERI, Japan)	100 MWth	Pressurized, 3.0MPa	Natural	Partially filled with water	Ishida et al. 2001
W17 (MRX, Ship Propulsion)	Ishida (JAERI, Japan)	100 MWth	Pressurized, 12.0MPa	Forced	Completely filled with water	Kusunoki et al. 2000
W18 (IRIS)	Carelli (Westinghouse, USA)	100-350 MWe	Pressurized, 15.5MPa	Forced	HP spherical with suppression pool	Carelli et al. 2001a, 2001b; 2001c and 2001d
W25 (“Daisy”)	Buongiorno (INEEL, USA)	50-150 MWe	Boiling, 7.4MPa	Natural	HP spherical, dry	N/A
W26 (MASLWR)	Modro (INEEL, USA)	35 MWe	Pressurized, with some boiling, 10.5MPa	Natural	Partially filled with water	N/A

Natural indicates full natural circulation, no pumps. *Forced* relies mainly on pumped flow.
However, even the forced circulation reactors have a significant degree of natural circulation.

through a NERI grant at the Idaho National Engineering and Environmental Laboratory (INEEL), University of Oregon and Bechtel Power Corporation.

In brief, organizations from the following countries have expressed interest in the IPSR concept, either as direct proposers or collaborators in the various on-going design projects, including: Argentina, Armenia, Brazil, Croatia, France, Italy, Japan, Korea, Mexico, Spain, the U.K., and the U.S.

W1.2. CONCEPT DESCRIPTION

Three subgroups can be identified within the IPSR reactor class:

1. Reactors with traditional pressurized water reactor (PWR) pressure and temperature operating conditions

2. Reactors with somewhat lower-pressure water coolant

3. Reactors with boiling water coolant.

A brief description of these three subgroups is presented in Sections W1.2-a, b, and c below, respectively. However, this categorization will not be used for evaluation of the potential for meeting the Generation-IV goals (see Section W1.3), i.e., the different IPSRs will be evaluated together. The developers of the concepts primarily wrote the concept summaries reported below. They have been edited for style and brevity. Some of their statements may not reflect the judgment of the Technical Working Group, which is reported instead in Section 3 of this appendix.

W1.2-a. IPSRs with Traditional PWR Operating Conditions (W10, W14, W17, W18)

These are small- or medium-size PWRs (30-350 MWe) with the reactor pressure vessel housing the whole primary system including the core and the core support structures, the steam generators, the pressurizer and the pumps. The steam generators are located in the annulus between the core barrel and the reactor pressure vessel wall. Both straight-tube and helical-tube steam generators are considered. A pressurizer with either active heaters and sprayers or passive control with or without nitrogen gas pressure is located in the reactor pressure vessel upper head. The control rods are inserted from the reactor pressure vessel top. Internal control rod drives are also being considered for some of the concepts. The smaller-size concepts rely on full-power natural circulation of the primary coolant while the larger-size concepts make use of canned-motor pumps or fully internal spool pumps while maintaining a relative large natural-to-forced-circulation flow ratio. The operating pressure ranges from 12 to 15MPa, the inlet and outlet temperatures range from about 270 to 330°C.

The core of these reactors is made of a modest number of PWR fuel assemblies with UO_2 or MOX fuel and modified pitch and fuel rod diameter. Some concepts adopt a triangular lattice. To maximize the irradiation cycle (up to 5 years) and to compensate for the loss of reactivity associated with the smaller-diameter core, the enrichment is slightly larger in most of the designs than in current LWRs (4 to 5% versus 3 to 4%). Most concepts adopt a single batch refueling strategy, with replacement of the entire core every 4 to 5 years, which reduces fuel handling as well as spent fuel storage requirements, but yields lower burnups and slightly higher fuel costs than in equal-length conventional cycles with partial refueling. Note that the single-batch long irradiation cycle is a common, but not an essential characteristic of these systems, which can also be operated with a conventional multi-batch refueling approach of intermediate length (i.e., 12–18 months). The use of diluted boric acid is eliminated in all concepts and long-term control of the core reactivity is performed mainly by means of the control rods and burnable poisons, e.g., gadolinium, erbium and boron. Because of the boron elimination, some designs feature alternative means to control the reactivity during cold shutdown and refueling.

W10—SMART

The SMART core is loaded with 57 UO_2 low enrichment square fuel assemblies. The core is designed for a fuel cycle length of about 3 years or longer with a relativity low core power density and no soluble boron. The reactivity control during the operation is achieved with 49 control rods that are connected to control element drive mechanisms with a fine maneuvering capability. The low core power density results in an ample critical heat flux margin that will accommodate any power transient and thus ensure the core thermal reliability during power operations. The soluble boron-free core concept inherently produces a strong negative moderator temperature coefficient over the entire fuel cycle.

Twelve identical once-through steam generators are located in the annulus formed by the reactor pressure vessel and the core support barrel. Each steam generator contains helical-coil Ti-alloy tubes wound around an inner shell. The primary coolant is on the outside of the steam generator tubes and, therefore, the tubes are under compressive loading, reducing the stress corrosion cracking and thus reducing the probability of tube rupture. The steam generators are modular and any defective steam generator can be replaced individually. The SMART pressurizer is an in-vessel self-controlled passive pressurizer located in the upper plenum and is filled with water, steam, and nitrogen gas. The primary system pressure is controlled by the partial pressure of nitrogen gas. To achieve forced circulation of the primary coolant, 4 pumps are installed vertically on the annular cover of the reactor pressure vessel. The pumps are canned motor type pumps that eliminate the problems of conventional seals and associated systems.

A schematic of the SMART primary system, safeguard vessel, and emergency systems is shown in Figure 1. The core decay heat during emergency situations is removed by a passive residual heat removal system. SMART has 4 independent residual heat removal trains with 50% capacity for each train so that operation of only two trains is sufficient to remove the decay heat. During a small LOCA the core is basically protected and covered by the large primary coolant inventory and the pressure balance between the primary system and the safeguard vessel surrounding the reactor pressure vessel. In addition, an emergency core cooling system (ECCS) is provided. The ECCS consists of 2 independent trains with 100% capacity for each train. The system can provide vessel refilling so that the decay heat removal system can function properly in the long-term recovery mode following the small LOCA event.

W14—CAREM

The CAREM prototype core has 61 hexagonal cross section fuel assemblies. The reactivity is controlled by means of Gd_2O_3 burnable poison in specific fuel rods and by movable absorbing elements belonging to the Adjust and Control System. Chemical compounds are not used for reactivity control during normal operation. The fuel cycle can be tailored to customer requirements, with a reference design

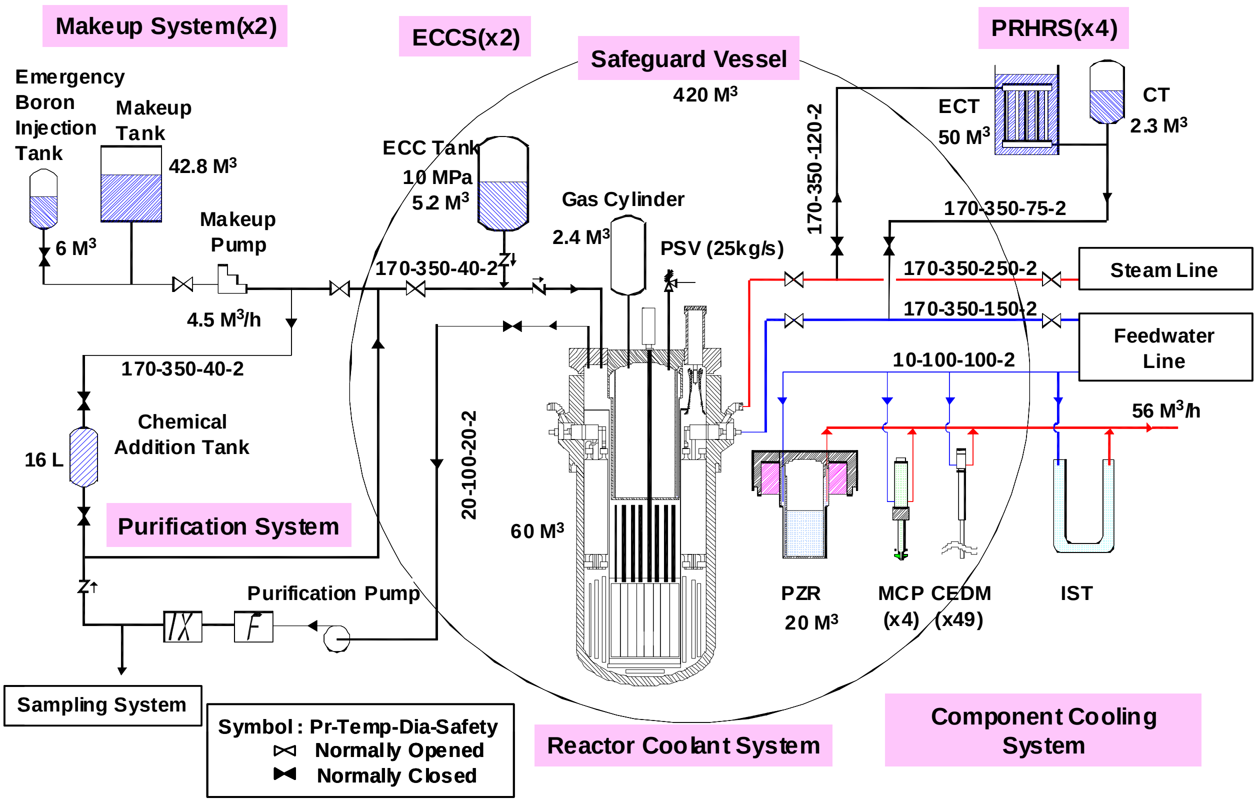

Figure 1. Schematic of the SMART primary system, safeguard vessel, and emergency systems.

of 330 full-power days and 50% core replacement. Twelve identical ‘mini-helical’ vertical steam generators, of the “once-through” type are placed equally distant from each other along the inner surface of the reactor pressure vessel. The location of the steam generators above the core produces natural circulation in the primary circuit. The secondary coolant circulates upwards within the tubes, while the primary goes in counter-current flow. The steam generators are designed to withstand the primary pressure without pressure in the secondary side, and the secondary system is designed to withstand the primary pressure up to the isolation valves for the case of steam generator tube break. Because of the self-pressurization of the reactor pressure vessel (steam dome) the system keeps the pressure very close to the saturation pressure. At all the operating conditions this has proved to be sufficient to guarantee a remarkable stability of the reactor pressure vessel pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in the case of power ramps.

A schematic of the CAREM containment and emergency systems is shown in Figure 2**.** The CAREM safety systems are based on passive features to mitigate accidents during a long time period and are duplicated to provide redundancy. The first shutdown system is designed to shut down the core when an abnormality or a deviation from normal situations occurs, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping a total of 25 neutron-absorbing elements into the core by the action of gravity.

The second shutdown system is a gravity-driven injection of borated water system at high pressure. It actuates automatically when the Reactor Protection System detects failure of the First Shutdown System or in the case of a LOCA. The system consists of two tanks located in the upper part of the containment, each connected to the reactor vessel by two piping lines: one from the steam dome to the upper part of the tank, and the other from a position below the reactor water level to the lower part of the tank. When the system is triggered, the valves open automatically and the borated water drains into the primary system by gravity. The discharges from a single tank will shutdown the reactor.

The residual heat removal system has been designed to reduce the pressure on the primary system and to remove the decay heat in case of a loss of heat sink. It is a simple and reliable system that operates by condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed. When the system is triggered, the outlet valves open automatically.

The emergency injection system prevents core damage in case of a LOCA. In the event of such accident, the primary system is depressurized with the help of the emergency condensers to less than 1.5 MPa, with the water level over the top of the core. At 1.5 MPa a low-pressure water injection system comes into operation. The system consists of two tanks with borated water connected to the reactor pressure vessel. The tanks are pressurized, thus when the pressure in the reactor vessel reaches 15 bar during a LOCA, the rupture disks break and start flooding the reactor pressure vessel.

Three safety relief valves protect the integrity of the reactor pressure vessel against overpressure. Each valve is capable of producing 100% of the necessary relief. The blow-down pipes are routed to the suppression pool.

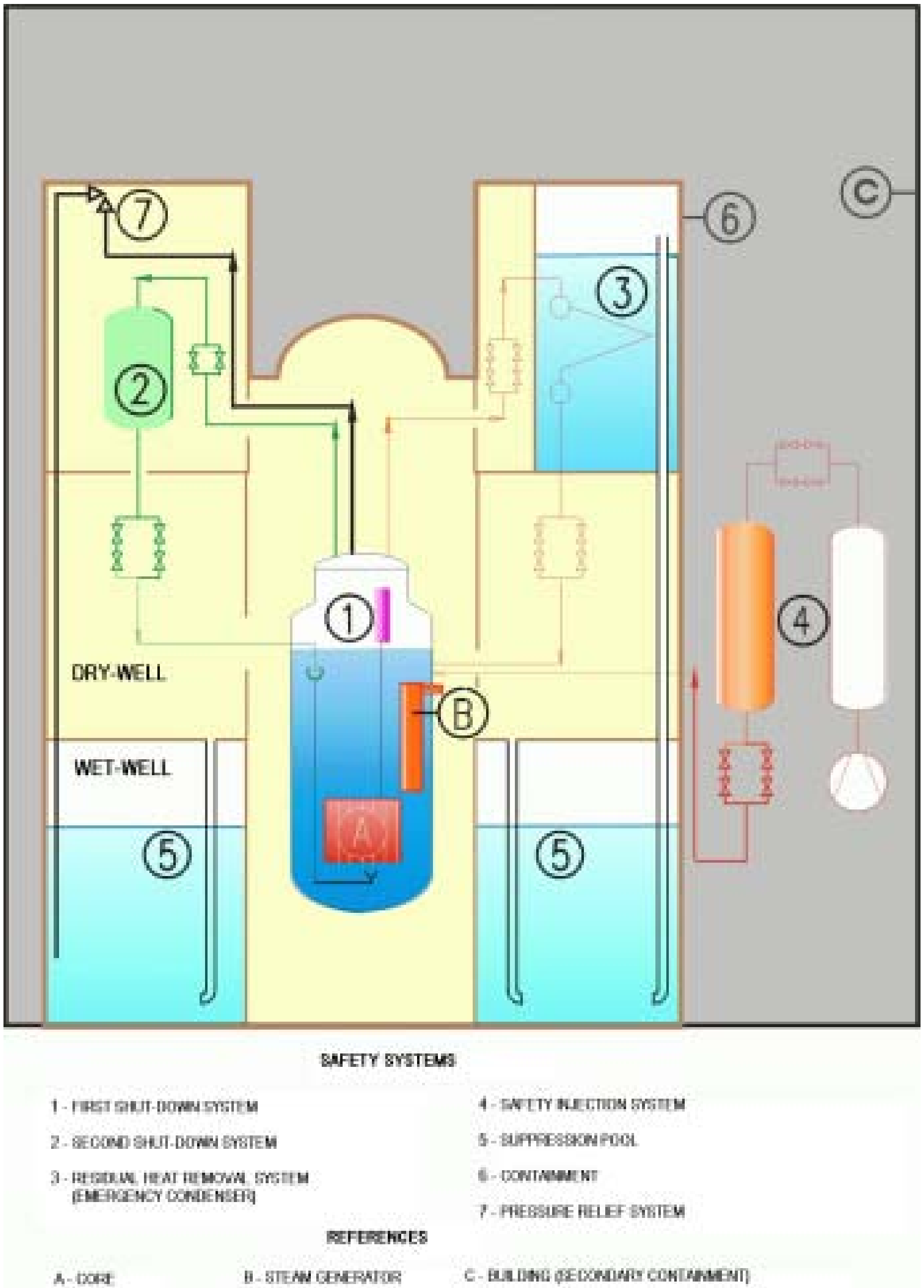

Figure 2. Schematic of the CAREM containment and emergency systems.

To increase proliferation resistance all the refueling tasks will be performed in the reactor hall, which is designed to allow remote monitoring of all the nuclear material handling. The entrance-exit and the interfaces have been designed to allow the counting of the items during movement. The entrance-exit has been designed to maximize the handling time compatible with operation needs. The Fuel Assemblies Pool could also be sealed and include remote seismic monitoring to detect a perimeter violation.

W17—MRX

This is an innovative and passively safe marine reactor system. The MRX has been designed to be lightweight and compact, and to enhance safety and reliability with the adoption of technologies such as an in-vessel type control rod drive mechanism, a water-filled containment, and a passive decay heat removal system. The MRX is a PWR of the integral-type with a 100 MW thermal output. Adoption of a small water-filled containment makes the MRX extremely lightweight and compact. The total weight and volume of MRX are about 1600 tons and 1210 m^3, respectively, roughly half those of the reactor in the first Japanese nuclear ship, *Mutsu*, although the MRX power is three times greater. The key parameters of the MRX are shown in Table 2.

Table 2. Main parameters of the MRX.

Reactor Power	100 MWt
Reactor coolant	
Operating pressure	12 MPa
Inlet/outlet temperature	282.5 / 297.5°C
Flow rate	4500 t/h
Core / Fuel	
Equivalent diameter	1.49 m
Effective height	1.40 m
Av. linear heat flux	7.9 kW/m
Fuel type	Zry-cald UO_2 fuel
^{235}U enrichment	4.3%
Fuel inventory	6.3 ton
Fuel Av. burn-up	22.6 GWd/t
Refueling interval	4 years (50% load factor)
Number of fuel assemblies	19
Fuel rod outer dia.	9.5 mm
Control rod drive mechanism	
Type	In-vessel type
Number of CRDMs	13
Main Coolant pump	
Type	Horizontal axial flow canned motor type
Rated power	200 kW
Number of pumps	2
Steam generator	
Type	Once-through helical coil type
Steam temp. / press.	289°C / 4.0 MPa
Reactor vessel	
Inner diameter / height	3.7 / 9.7 m
Containment	
Type	Water-filled RV immersion type
Inner diameter / height	7.3 / 13 m
Design press.	4 MPa

The water-filled containment encloses the reactor pressure vessel for prevention of radioactive material release to the surroundings. Core flooding can be maintained passively by the pressure balance between the containment and reactor pressure vessel during the early part of a LOCA, and, in the later period of the transient, with the help of the emergency decay heat removal system and the emergency containment water cooling system.

W18—IRIS

The IRIS modules have a power range between 100 and 350 MWe. The IRIS fuel cycle is a long life, straight through burn fuel cycle, without fuel shuffling or partial refueling. The 5-year first core has 4.95% enriched UO_2 fuel in a square pitch configuration almost identical to current Westinghouse PWR assemblies. This was selected for ease of licensability. Reload cores will have an 8 to 10 years lifetime with higher enriched (8–10%) UO_2, or with MOX fuel, and will reach a peak burnup of 90,000 MWd/ton.

The layout of the IRIS primary coolant vessel is shown in Figure 3. IRIS has eight steam generators and eight totally immersed spool-type pumps. The steam generators are of the helical type, which has been adopted in previous water and sodium cooled integral reactors. Integral shields of borated carbon steel significantly reduce the dose at the vessel surface. IRIS has emphasized the “safety by design” concept where accidents are eliminated by design (e.g., large LOCAs because of integral configuration, other LOCAs because of the containment design) or at least their consequences and probability are greatly reduced.

IRIS adopts a small, spherical high-pressure containment to basically eliminate the consequences of small-to-medium LOCAs (which are historically the accidents yielding the worst consequences). The water inventory within the reactor pressure vessel after a LOCA is maintained by reducing the pressure differential between the vessel and containment, thus reducing the driving force across the rupture and ultimately the coolant loss. This is accomplished through (1) the high-pressure spherical containment, which increases the pressure downstream of the break and (2) an efficient heat removal system inside the vessel which reduces the pressure upstream of the break. The integral vessel design enables reduction of the containment size to about half the diameter needed in a comparable LWR. Thus, at the same stress level in the metal shell, the spherical containment can take a pressure at least four times higher than the cylindrical containment in a loop reactor. The multiple steam generators remove heat from inside the vessel. In addition, the large water inventory inside the vessel provides a grace period by slowing the transient evolution. Because the LOCA is not a serious concern with this design, IRIS does not have an emergency core cooling system (ECCS).

The safety by design approach is not limited to LOCAs, rather, the entire accident spectrum (loss of flow, steam and feed-water breaks, steam generator tube ruptures, station blackout, etc.) are addressed. Initial evaluations indicate that out of the eight Class IV accidents considered for the AP-600 reactor design, seven are either eliminated or downgraded to Class III or lower and the only remaining (refueling) accident has a much-reduced probability of occurring.

IRIS is designed to have maintenance shutdown intervals of 4 years or greater to match the refueling interval and decrease operation and maintenance costs. In-vessel components, diagnostic and maintenance are designed to achieve this goal; this represents a major research and development item.

Preliminary analyses have indicated that the IRIS cost of electricity, estimated at or below 3¢/kwh, is competitive with all power options.

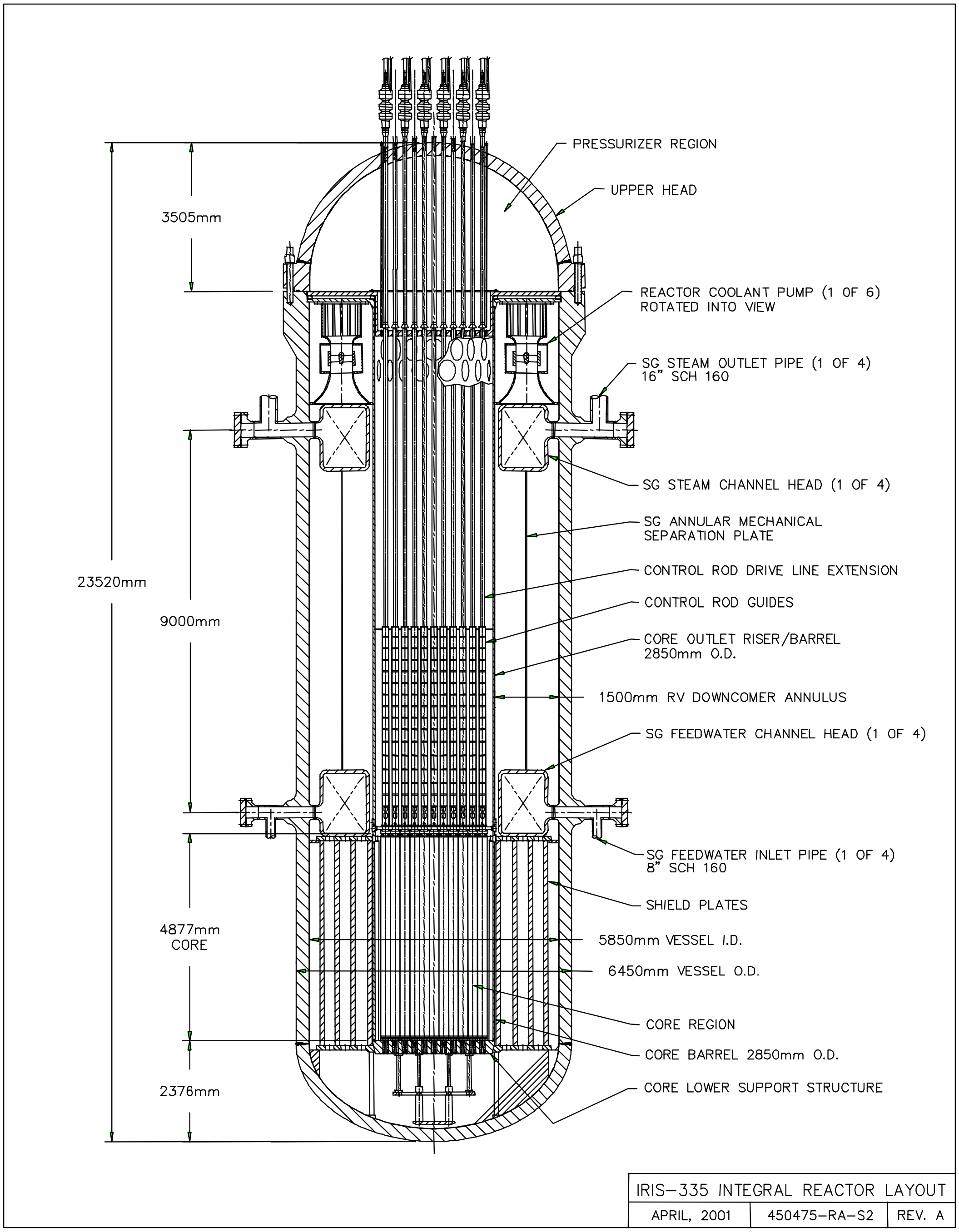

Figure 3. IRIS vessel layout (300 MWe).

W1.2-b Small IPSRs with Low-Pressure Water Coolant (W16, W26)

These are small-size (<100MWe) pressurized water reactors whose operating pressure and temperatures are reduced to improve safety (i.e., smaller accumulated energy, larger safety margins) and simplify the plant (i.e., reliance on fully passive emergency systems). Some coolant boiling is allowed in the Multi-Application Small Light Water Reactor (MASLWR) design. High capacity factors are pursued by increasing the irradiation cycle (up to 10 years) and by adopting full-power natural circulation for greater reliability. Because of the lower operating conditions, the thermal efficiency of these plants is relatively low (<30%).

W16—Passively Safe Small Reactor for Distributed Energy Systems (PSRD)

The PSRD is suitable for barge or underground siting, where the reactor is entirely fabricated in a factory, and then shipped and operated on a barge or placed underground. It is an integral type light water reactor with natural circulation and self-pressurization in the primary system and a low-to-medium power, 50 to 500 MW thermal. A wide fuel rod pitch is used to enable full-power natural-circulation operation. One of design goals is to achieve a long core life, about 10 years without fuel shuffling and refueling, using low enriched (less than 5%) UO_2 fuel. Soluble boron is not used for reactivity control. The PSRD uses a straight tube type steam generator. Major parameters for a PSRD with a 100 MW thermal output are presented in Table 3. A conceptual drawing of the PSRD is shown in Figure 4.

Table 3. Main parameters of the PSRD.

Reactor Power	100 MWt
Reactor coolant	
Coolant	Natural circulation and self pressurized
Operating pressure	3 MPa
Outlet temperature	233°C
Core / Fuel	
Equivalent diameter	1.6 m
Effective height	1.5 m
Average linear heat flux	7.3 kW/m
Fuel type	Zry-cald UO_2 fuel
^{235}U enrichment	less than 5%
Fuel inventory	6.6 ton
Fuel Av. burn-up	28 GWd/t
Core life	10 years (50% load factor)
Number of fuel assembly	37
Fuel rod outer diameter	9.5 mm
Control rod drive mechanism	
Type	In-vessel type
Number of CRDM	37
Steam generator	
Type	Once-through helical coil type
Steam temperature	180°C
Reactor vessel	
Inner diameter / height	4.0 / 12.0 m
Containment	
Inner diameter / height	8.5 / 22 m

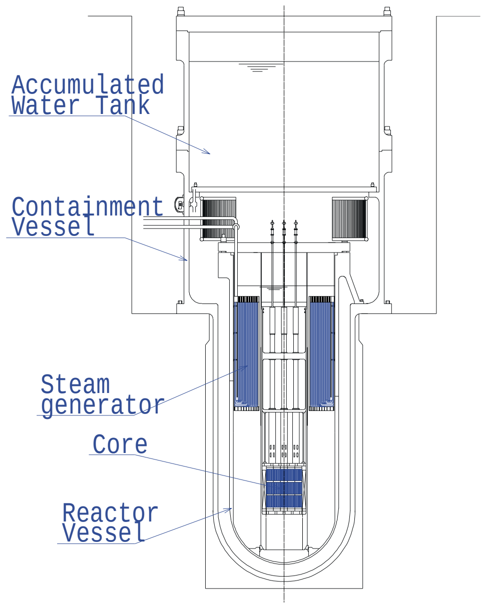

Figure 4. The PSRD primary system and containment.

When the PSRD is installed at a depth of 50 m under an energy consumption area, it will be automatically and remotely operated from a supervisory and control center. In the case of a reactor accident, immediate public evacuation will not be necessary because a significant environmental impact at the ground surface will not appear for several decades after the accident.

W26—Multi-Application Small Light Water Reactor (MASLWR)

A MASLWR module consists of an integral reactor/steam generator located in a steel cylindrical containment. The entire module is 4.3 m (14 ft) diameter and 18.3 m (60 ft) long. This size allows it to be entirely shop fabricated and transported to site on most railways or roads. Two or more modules are located in a reactor building, each being submersed in a common, below grade cavity filled with water (Figure 5). Cooling of the containment during normal and abnormal conditions is accomplished by heat

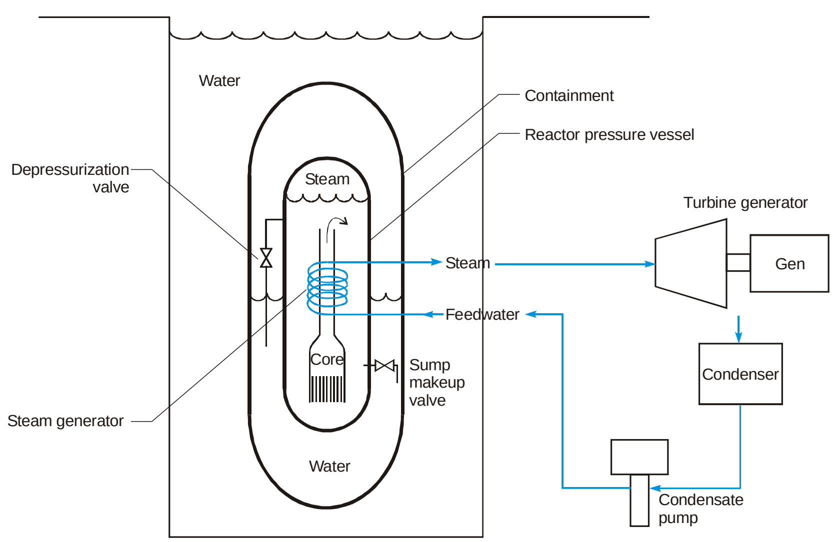

Figure 5. Schematic of the MASLWR reactor.

conduction through the containment steel walls to the cavity water. Heat from the cavity water is removed through a closed loop circulating system and rejected into the atmosphere in a cooling tower to maintain a pool temperature below 38°C. For the most severe postulated accident, the volume of water in the cavity provides a passive ultimate heat sink for three or more days allowing the restoration of the lost normal active heat removal systems.

The core is located in the lower part of the vessel, with the steam generator above it. To effectively achieve full-power natural circulation, the core is connected directly to the space above the heat exchanger via a large-diameter tube, or riser, which is an upper extension of the core barrel.

The core consists of standard PWR assemblies, with an active fuel height of approximately 1m. The overall height to diameter ratio for the core is approximately 1. The fuel is cylindrical-pins with a cladding outer diameter of 9.5 mm, and a pitch-to-diameter ratio of 1.33. The fuel pellets are UO_2, enriched to <20% U-235.

The steam generator is a helical-tube, once-through heat exchanger. The heat exchanger consists of approximately 1000 tubes, arranged in an upwardly spiraling pattern. Cold feed water enters the tubes at the bottom, and saturated steam is collected at the top. The heat generated in the core raises the temperature of the primary coolant from 191 to 314°C. This heat is removed while flowing down across the steam generator tubes.

Safety performance of the MASLWR unit is based on a typical ALWR fast shutdown system, two independent automatic depressurization system (ADS) trains, and heat transfer from the containment. In emergencies the reactor is scrammed and the ADS activated. The pressures in the primary system and the containment are quickly equalized (the containment is designed for 1.7MPa), and the ADS flow paths assure natural circulation of the coolant between the containment and the primary system providing cooling to the core. The containment itself is completely submerged in a large pool of water, which serves as the ultimate heat sink for cooling the reactor.

W1.2-c. IPSRs with Boiling Water Coolant (W25)

W25—"Daisy"

This reactor is basically a small-size (<150 MWe) natural-circulation BWR with an indirect cycle and a fully passive decay-heat removal system (see Figure 6). The reactor operating temperature and pressure are 290°C and 7.4MPa, respectively. The steam generated in the core is condensed in condensing units located within the steam dome at the top of the pressure vessel. A key feature of this reactor is that the secondary water (flowing in the condensing unit tubes) is maintained liquid at a pressure higher than the primary system pressure (8.0 MPa) so that if a tube rupture occurs, there is no

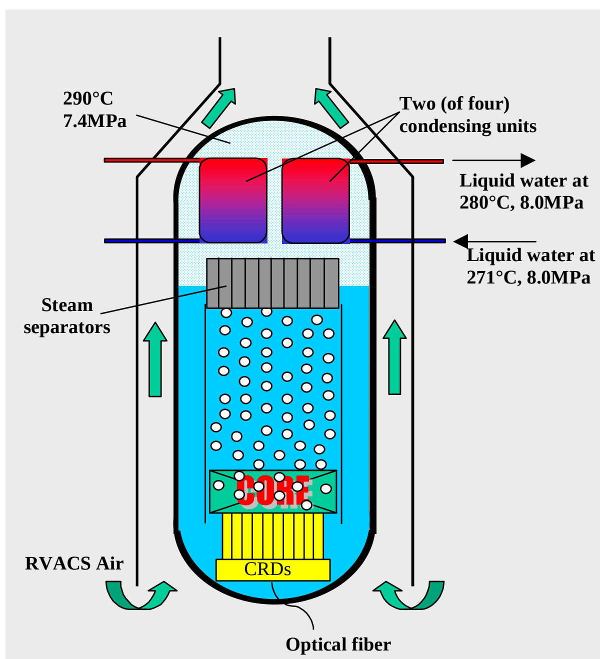

Figure 6. Schematic of Daisy's primary system and RVACS.

release of the primary coolant. Therefore, to generate steam, the flow of secondary water must be subjected to an abrupt and large pressure drop in a dedicated throttling valve that causes some water to flash to steam. This steam is dried in a moisture separator and then is sent to the turbine (throttle conditions: a temperature of 275°C, pressure of 6.0 MPa). From this point on, the power cycle is similar to that of traditional LWRs. Thermal efficiencies up to 29% are possible, somewhat smaller than typical LWRs because of the large pressure drop in the throttling valve.

The core is made of 205 off-the-shelf BWR fuel assemblies of the 8 x 8 type. The core equivalent diameter is 2.5 m. The fuel is uranium dioxide with an enrichment of about 3%, typical of current BWRs. Higher enrichments are not needed because the reactivity loss from the smaller-diameter core is more than compensated for by the lower void fraction in the core (10 versus 30%). The average power density of the core is about three times smaller than traditional BWRs. Therefore it is possible to operate the core for 4 to 5 years without refueling. Also, the core exhibits much larger margins to CHF and onset of density-wave instabilities than typical BWRs. In the reference configuration the control rod drives are located within the vessel and controlled by a single optical fiber cable.

During a station blackout or other loss-of-normal-heat-sink events the decay heat is removed by an RVACS-type system located directly on the outer surface of the reactor pressure vessel. The reactor vessel auxiliary cooling system (RVACS) is based on atmospheric air that flows in a 23 cm annulus around the vessel (the so-called collector). The heat is transferred through the vessel by conduction and then discharged to the air by convection and radiation.

W1.3. POTENTIAL FOR CONCEPT MEETING GENERATION IV GOALS

In the following sections, the IPSR concept set is assessed against the Generation-IV goals. The advantages and/or disadvantages of the IPSR concept set are evaluated relative to a typical Generation-III reactor (e.g., the AP-600, ABWR, and System80+ designs), which serve as the reference system. In those areas for which no appreciable differences can be identified between the IPSR concept set and the reference, the analyzed concept is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and metrics by means of a label in parenthesis.

W1.3-a. Evaluation Against High Level Criteria

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

IPSRs exhibit the following advantages in the area of natural resource utilization:

- Because some of these reactors operate at lower power density and thus lower fuel temperature than current LWRs, it may be possible to achieve somewhat higher burnup levels resulting in more electric energy and less radioactive waste generated per unit mass of natural uranium. (SU1-1)

- If plutonium-based MOX fuels are utilized, it is possible to significantly increase the amount of electric energy generated per unit mass of natural uranium. [Note: the use of MOX fuel is not unique to IPSRs.] (SU1-1)

On the other hand, because these are thermal reactors, plutonium breeding is not possible and thus the utilization of natural uranium resources is limited compared with fast reactors.

It is concluded that IPSR systems are substantially equivalent to the reference LWRs in the area of fuel utilization.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

IPSRs exhibit the following advantage in the area of waste minimization:

- For most concepts, the use of borated water as a long-term reactivity control means is eliminated, which results in less waste. [Note: the use of boron-free water chemistry is not unique to IPSRs.] (SU2-1)

- IPSRs have the following disadvantage in the area of waste minimization:

- For given installed capacity, multi-module plants are expected to have more activated materials (such as in-pile structures and instrumentation) than large monolithic plants. (SU2-1)

Because spent fuel is the radioactive waste of greatest concern, it is concluded that IPSR systems are substantially equivalent to the reference LWRs in the area of waste minimization.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

IPSRs exhibit the following advantages in the area of proliferation resistance:

- The use of low-enriched uranium and the lack of recycling makes diversion of IPSR UO2 fuel a relatively unattractive path to proliferation. (SU3-1)

- The potentially higher burnup achievable with IPSR oxide fuel would yield end-of-life plutonium isotopics rich in non-fissile isotopes and relatively poor in Pu-239. (SU3-1)

- The long in-pile residence time minimizes the opportunity for plutonium diversion. (SU3-1).

The last two barriers to proliferation can be bypassed by extracting the fuel early in the irradiation cycle; however, this would be relatively transparent.

IPSRs exhibit the following disadvantages in the area of proliferation resistance:

- The long in-pile residence times and smaller core sizes require somewhat higher fuel enrichments (only slightly greater in most of these designs, up to nearly 20% in one design) increasing the front end fuel cycle proliferation risk (enrichment and conversion facility, fuel transportation and handling).

- Some IPSRs are designed for MOX recycle. If conventional wet processes are used to separate the fissile material, there will be an increased potential for weapons material diversion.

It is concluded that in terms of proliferation resistance, the IPSR concepts are slightly better than the reference LWRs.

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

IPSRs exhibit the following advantages in the area of safety and reliability under normal operating conditions:

- Because they operate at lower power density than current LWRs, the margin to CHF at steady-state conditions and during transients is larger. (SR1-2)
- Lower temperatures in the fuel may result in better fuel reliability as well as a lower release of fission gases upon fuel pin failure. (SR1-2, SR1-3)
- In the full-power natural circulation system designs, pump trips are eliminated as accident initiators. (SR1-3)
- Use of an indirect cycle reduces the probability of release of radioactivity from the plant. (SR1-1).

IPSRs have the following disadvantages in the area of safety and reliability under normal operating conditions:

- Monitoring, inspection, and maintenance of the in-vessel primary system components may be difficult. Innovative solutions will be required, whose impact on the overall plant reliability is uncertain at this point. (SR1-3, SR1-1)
- For given power output, a multi-modular plant is likely to have more components (e.g., pumps, SGs, valves) than a large monolithic plant. (SR1-3)

The evaluators believe that the IPSRs have the potential to perform better than the reference LWRs in terms of safety and reliability under normal operating conditions. However, at this point large uncertainties exist associated with the issue of monitoring, inspecting and maintaining the in-vessel primary-system components.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

IPSRs exhibit the following advantages in the area of safety and reliability under accident conditions:

- Large LOCAs are eliminated by design. (SR2-3)
- If in-vessel control-rod-drive mechanisms are adopted, the possibility of a control-rod-ejection accident is eliminated for most of the IPSR concepts. (SR2-3)
- For those concepts with the primary coolant on the shell side of the steam generators, the possibility of a steam generator tube rupture (e.g., by stress-corrosion cracking) is greatly reduced, because the tubes are in compression. (SR2-3)

- In those designs with a small high-pressure containment thermodynamically coupled to the vessel, the consequences of small LOCAs are mitigated without the need for water injection in the core. (SR2-3)

- In the design with the secondary system at higher pressure than the primary system, the probability of small LOCAs associated with a tube rupture is greatly reduced. (SR2-3)

- In the full-power natural circulation system designs, LOFAs are eliminated. In the forced-circulation system designs with a high degree of natural circulation, the consequences of LOFAs are greatly mitigated. (SR2-1)

- Upon loss of the normal heat sink (e.g., the steam generators) removal of the decay heat from the core is provided by redundant, passive systems that do not require any intervention by the operator to protect the core for a relatively long period. (SR2-1)

It is concluded that the IPSR concepts will perform significantly better than the reference LWRs in terms of safety and reliability under accident conditions.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

IPSRs exhibit the following advantages in the area of severe accident mitigation and need for offsite emergency response:

- IPSRs have a large primary-coolant inventory per unit thermal power. (SR3-1)

- Some designs have higher-pressure containments. (SR3-1)

- For those concepts with the primary coolant on the shell side of the steam generators or with the secondary system at higher pressure than the primary system, the possibility of containment bypass due to a steam generator tube rupture is greatly reduced. (SR3-1)

It is concluded that the IPSR systems will perform better than the reference LWRs in the area of severe accident mitigation and need for offsite emergency response. It should be noted that IPSRs are designed to enhance safety throughout the spectrum of postulated accidents.

Economics–1. Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

IPSRs exhibit the following advantages in the area of operating costs:

- The anticipated longer irradiation cycles and the potentially higher reliability of the primary systems should result in higher plant capacity factors and significantly reduced maintenance costs. Some concepts design the reactor for a 4 or 5-year maintenance interval to match the refueling interval. (EC-3)

- In the low power IPSRs, the use of natural circulation eliminates pump maintenance. (EC-3, EC-4)

- The high degree of natural circulation in the higher power IPSRs reduces the pumping requirements. (EC-3, EC-4)

IPSRs have the following disadvantages in the area of operating costs:

- Operation and maintenance of many reactor modules at a single site might result in higher operation and maintenance costs than in current LWRs because of the increased number of components, control rooms, etc. (EC-3)

- To achieve a longer irradiation cycle, some IPSR concepts make use of slightly more enriched uranium than current LWRs. Also, a longer irradiation cycle increases the carrying charges on the fuel. Therefore, the cost of the fuel per unit electric energy generated is expected to be somewhat higher. (EC-3)

- Those IPSR concepts with lower thermal efficiency will also have an even higher fuel cost per unit electric energy generated. (EC-3)

- Because the primary system is contained within the pressure vessel, online inspection and maintenance of the components (e.g., steam generators, pressurizer, pumps) might be more costly and could cause unplanned, lengthy outages. (EC-3)

At this point the evaluators believe that it is possible that IPSRs will perform better than the reference LWRs in terms of operating costs, but better cost analysis must be performed once the newest designs are completed and their capacity factors evaluated.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

IPSRs exhibit the following advantages in the area of capital costs and financial risk:

- The nuclear island is greatly simplified by eliminating the external-loop piping and many safety-grade systems. (EC-1)

- The design and fabrication approach for the IPSRs is based on modularity:

  - The reactor modules can be fully fabricated in a factory and be readily transported to the site, which reduces expensive on-site assembling/welding and, ultimately, construction time. (EC-1)

  - If a relatively large number of reactor modules will be needed, it will be possible to take full advantage of cost reductions due to learning and standardization. (EC-1)

  - Additional generating capacity can be gradually installed by adding small modules; this will allow the production to match the electricity demand of the utility customers, prevent market saturation, and ultimately maintain a stable price of electricity. (EC-2)

  - For large plants with many reactor modules, it will be possible to put the first few reactor modules into operation relatively quickly and generate an early cash flow. (EC-2)

  - The initial cost outlay is significantly less than current LWRs.

  - Due to the lower plant power and lower funding needed for IPSRs, the financial risk is lower.

- Smaller (and for some designs no) pumps will be required because of the high degree of natural circulation. (EC-1)

- Because the power density is smaller than current LWRs, the damage to the vessel from fast neutrons should be modest. (Also, some designs feature a relatively wide down-comer and internal shields that further reduce the irradiation damage to the vessel.) Therefore, it is expected that the reactor lifetime can be extended well beyond that of current LWRs. (EC-5)

IPSRs have the following disadvantages in the area of capital costs and financial risk:

- The smaller power per reactor module and smaller power density within the core may result in a larger plant size and amount of materials per unit power generated. (EC-1)

- For a given electric power output, a plant with many reactor modules likely has a larger footprint than a plant with a single large monolithic reactor. (EC-1)

- The reactor vessel is larger and thicker than loop-type LWRs on a per MWe basis. (EC-1)

The evaluators believe that at this point it is not possible to predict with certainty whether IPSRs will perform better than the reference LWRs in terms of capital costs and financial risks. Moreover, the type of market is extremely important. For example, many developing countries cannot accept large plants because of electric grid conditions and local power requirements. However, some highly industrialized countries may need larger modules.

W1.3-b. Summary of the Strengths and Weaknesses

A. Strengths of the IPSR concepts include:

- Simplification of the nuclear island

- Higher flexibility in meeting the needs of the electric grid

- Potential for higher capacity factors

- Elimination of large LOCAs

- Elimination of small-to-medium LOCAs as a safety concern (in some designs)

- Elimination/mitigation of LOFAs

- Elimination of rod ejection accidents (in some designs)

- Passive removal of the decay heat under accidental conditions

- Small or no pumps required

B. Weaknesses of the IPSR concepts include:

- More difficult inspection of the primary system

- Smaller power density than in current LWRs (for most designs)
- Larger plant footprint for a given installed capacity
- Slightly higher fuel costs
- Thermal efficiency at or below current LWR levels

W1.4. TECHNICAL UNCERTAINTIES

W1.4-a. Research Needs

The following research needs have been identified regarding the IPSR concept:

1. Evaluation of the economic viability (i.e., capital and operation and maintenance costs, financial risk, etc.) of the modular design approach
2. Design and demonstration of the in-vessel primary-system components (steam generators, pumps, etc.)
3. Corrosion control for the fuel cladding throughout the long irradiation cycles
4. Development and/or qualification of in-vessel control-rod-drive mechanisms
5. Development and qualification of equipment designs for infrequent maintenance
6. Demonstration of the higher burnup levels potentially achievable for oxide fuel operating at somewhat lower temperatures
7. Probabilistic risk analysis to show that, for given installed capacity, the core damage frequency and dose distribution of a multi-module plant is significantly smaller than that of a single-reactor plant, to eventually show that emergency response is not needed.

W1.4-b. Institutional Issues - Licensability & Public Acceptance

No new and/or specific public acceptance issues were identified for the IPSR concept. This concept set is best characterized as an evolutionary design. The public should be receptive to the elimination of accident initiators by design, to the superior passive safety performance of these systems, and to the possibility of eliminating the need for emergency response.

Licensing of these reactors should be made easier by maximizing the use of existing LWR technology, i.e., fuel, materials and equipment. For those components or systems that are new, it will be necessary to conduct supporting experiments to demonstrate their performance and reliability. It will also be necessary to demonstrate that the state of the in-vessel primary-system components can be properly monitored during operation.

W1.4-c. Timeline for Deployment

Given the relatively small R&D requirements for these reactors, it is expected that the IPSR concepts could be considered for early deployment (before 2015).

W1.5. INITIAL ASSESSMENT: OBSERVATIONS AND CONCLUSIONS

The integral primary-system reactor concepts are good candidates for further assessment. At this point the key issues that will emerge for determining the relative ranking of these systems appear to be the economic viability of a modular design approach as well as the reliability and design of the in-vessel components.

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W1-7. TOP-TIER SCREENING SHEET—INTEGRAL PRIMARY-SYSTEM REACTORS

Summary Evaluation: __X__ Retain ____ Reject

	Goal	- -	-	+	+ +	Comments
SU1	Fuel Utilization		E			Thermal reactors with about the same enrichment and burnup as current LWRs
SU2	Nuclear Waste		E			Thermal reactors with about the same enrichment and burnup as current LWRs
SU3	Proliferation Resistance					Long irradiation cycle with no shuffling; modular core: can take back fuel to the vendor
S&R1	Worker Safety and Reliability					1st of a kind: performance and maintenance of the in-vessel components; larger number of components than current LWRs for given power output; indirect cycle minimizes worker exposure Nth of a kind
S&R2	CDF					Most accident initiators eliminated or mitigated, passive safety systems
S&R3	Mitigation					Large inventory of water per unit power. LOCAs, LOFAs, and other accidents eliminated or mitigated.
E1	Life-cycle cost					1st of a kind: higher fuel cost; reliability is an issue Nth of a kind: long irradiation cycle and maintenance interval results in high capacity factor
E2	Financial Risk/Capital Cost					1st of a kind: small power per module; for given power output, more components and materials Nth of a kind: simplified nuclear island; smaller initial investment; can follow grid demand

Appendix W1: Integral Primary System Reactors

Appendix W2

Loop Pressurized Water Reactors Concept Set Report

December 2002

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ABSTRACT

This set comprises two reactor concepts that are best described as loop pressurized water reactors (PWRs). The essential innovative characteristic of these reactors is the use of a safeguard vessel that envelopes the whole primary system (i.e., the main pressure vessel, steam generators, control rod drives, and pressurizer) for mitigation of primary system component failure. The coolant/moderator is pressurized light water. The primary-coolant mode of circulation is forced. The proposed concepts are thermal reactors. One concept makes use of low-enriched-uranium oxide-fuel, clad with Zircaloy while the other uses thoria-urania fuel within a zirconium metal matrix with the fuel rods arranged in a closed packed hexagonal array. These concepts use less or even no boron for reactivity control compared to traditional PWRs.

The proposed concepts try to maximize the use of existing light water reactor technology and passive systems to cope with the consequences of accident events. Moreover, the adoption of the additional vessel enables elimination of some safety systems.

This reactor concept offers potential for superior safety compared with the reference LWRs. However, issues to be resolved include reliability and maintenance of the primary system components that are not easily accessible, and impact of the additional vessel on the capital cost, even though some safety systems are eliminated.

The concept is retained for further assessment.

W2.1. INTRODUCTION

This set comprises two reactor concepts. One is the "Simple & Intelligent PWR with Bloc Type/Double Vessel Utilizing Compact Thoria-Urania Dispersed Metal Fuel" (Bloc); the other is the "Multipurpose Advanced Reactor, Inherently Safe" (MARS). The common innovative characteristic of these reactors is the use of a safeguard vessel that envelopes the whole primary system (i.e., the main pressure vessel, steam generators, control rod drives, and pressurizer) for mitigation of primary system component failure. However, significant differences exist.

The Bloc reactor is a large pressurized water reactor (PWR) with an electrical output >1,500MWe and the MARS is a small PWR (150MWe). The Bloc PWR operates at typical PWR pressures and temperatures, while MARS operates at substantially lower temperatures and pressures for reduction of the structural materials oxidation and reduction of the energy accumulated in the primary system. Some design features of the Bloc Type PWR are revolutionary compared to the reference ALWR. However, the concept builds on the Korean ALWR designated as the APR1400 (Advanced Power Reactor, 1400MWe) that is currently in the final stage of development and is to be in commercial operation in 2010 in Korea. The design features of the MARS reactor are more evolutionary. The MARS project started in 1983 with the objective of developing a reactor to be used for a wide range of applications, including desalination and district heating. The MARS design was developed over 15 years, and the proponents claim it is almost ready for deployment after minor verification/validation of its engineering features. The MARS would be adequate for deployment in countries with a need for small-to-medium-size plants.

The general characteristics of these two reactors are compared in Table 1.

Table 1. General Characteristics of the Loop PWRs

	Bloc-Type PWR	**MARS**
Gen-IV Designation	W11 (Bloc Type PWR)	W3 (MARS)
Proponent	Park (KAERI, Korea)	Sorabella (Univ. of Rome, Italy)
Power (MWe)	>1500	150
Thermal Efficiency	35%	25%
Coolant/Pressure	Light water, pressurized, 15.0MPa	Light water, pressurized, 7.5MPa
Circulation Mode	Forced	Forced
Fuel	Thoria-Urania dispersed in Zr Metal	LEU oxide
Cycle Length	10 years	18 months
Decay Heat Removal	Passive (air on the outer containment surface)	Passive (LP emergency condensers)
Special Features	Safeguard vessel around the primary system	Double-walled primary system
Safety Features	LOCAs and severe accident mitigated	LOCAs and severe accident mitigated

W2.2. CONCEPT DESCRIPTION

A brief description of the two reactor concepts is reported in Sections W2.2a and W2.2b below. However, the evaluation of the potential for meeting the Generation IV Goals will be done as a concept set in Section W.3. The developers of the concepts primarily wrote the concept summaries reported below. They have been edited for style and brevity. Some of their statements may not reflect the judgment of the Technical Working Group, which is reported instead in Section 3 of this appendix.

W2.2-a. Large PWR (Bloc-Type PWR)

The design features, which are not described specifically in the concept report, are the same as those of the Korean APR1400.

Figure 1 shows a schematic diagram of the Bloc Type/Double Vessel PWR. The reactor coolant system (RCS) adopts a bloc-type double vessel arrangement, where the primary components are directly inter-connected. The primary coolant system consists of the reactor pressure vessel, steam generators, a pressurizer, and reactor coolant pumps. The steam generator nozzles are connected directly to the pressure vessel nozzles. There is no primary system piping. The number and type of steam generators are yet to be determined.

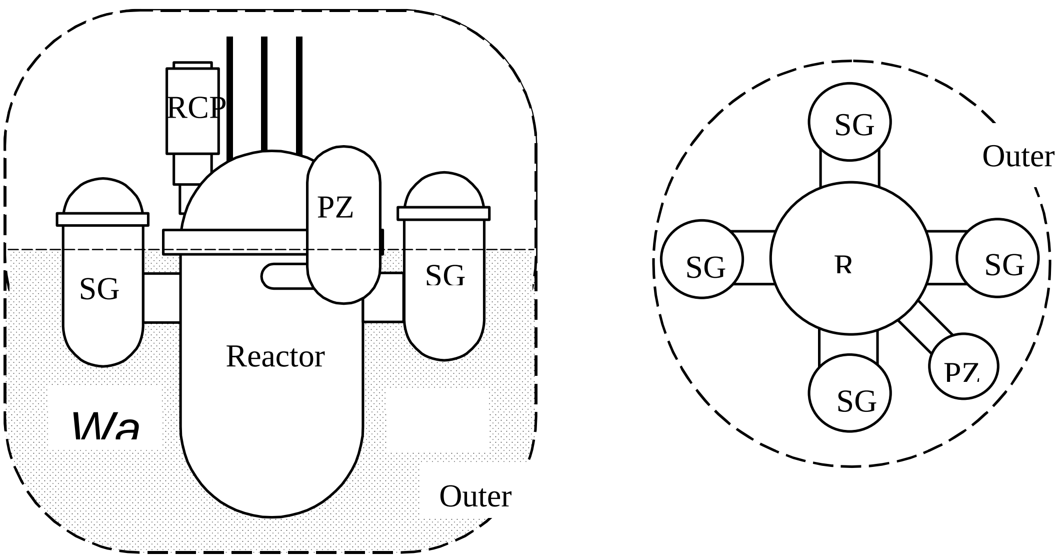

Figure 1. Bloc-type double vessel arrangement. The outer vessel in Option 1 covers the entire primary system and in Option 2 covers part of the primary system**.**

The primary system is encapsulated, partially or fully by a secondary shell (outer vessel) to form a so-called 'double vessel system.' The outer vessel is filled with water at a pressure much lower than the primary system. The outer vessel performs the function of shielding, and provides a water source to be injected passively into the primary system in case of an emergency situation. Therefore, the double vessel system eliminates the need for a conventional safety injection system. However, a traditional containment is also used to protect against external events and provide additional safety.

The proponents of the concept estimate that with the use of thoria-urania metal-matrix-based fuel, the reactor could operate for 10 years without refueling and achieve high burnup(>90,000MWd/t) The fuel would consist of thoria-urania dispersed in zirconium metal. This type of fuel has a higher thermal conductivity than the ceramic type fuel used in the reference ALWRs. The zirconium metal matrix material can suppress the fission gas release during both normal operation and accident conditions. The fission products in the high-level waste will also be retained, if the waste retains its fuel assembly structure.

Passive systems are provided for the removal of the decay heat. A core catcher in combination with the flooded reactor vessel will assure the retention inside the reactor vessel of any molten fuel that develops during a severe accident.

Obviously, the reactor safety and economic performance could greatly benefit from general advancements in nuclear engineering that however would not be unique to this reactor. These improvements include the development of online fully-automatic I&C systems, optic-fiber and/or wireless I&C systems, maintenance-free components, in-vessel control rod driving mechanisms, and nano-particles to enhance the heat transfer between the fuel cladding and the water coolant. All these things are being considered by the Bloc project in Korea.

W2.2-b Small PWR (MARS)

MARS is a small advanced PWR developed in Italy. It generates about 150MWe, with a modular solution to satisfy progressively increasing power requirements from the station. The MARS core is made of 89 standard PWR fuel assemblies. Less boron is used in the primary coolant compared with the traditional PWRs. The core is equipped with a passive shutdown system in addition to the standard shutdown system.

Figure 2 shows the primary cooling system of the MARS. The primary cooling system has only one loop, with 25” ID pipes, one canned pump, and one vertical axis U-tube steam generator. The pressure of the primary cooling system is lower (7.5MPa) than traditional PWRs. Every component of the primary system is encapsulated by an outer shell (pipes around pipes and vessels around vessels) filled with high-pressure water to eliminate the possibility of a loss of primary coolant in case of failure of the primary system.

The operating temperature and the thermal efficiency of the MARS reactor are lower than the traditional PWRs (i.e., 229^{o}C and 25%, respectively). The lower operating temperature and pressure of the primary coolant significantly reduce the oxidation of all the materials in the core and also reduce the energy accumulated in the primary system. The proponents expect that contamination of the primary coolant will be very low compared with traditional PWRs.

The proponents of the MARS concept claim the bus-bar cost of electricity from their plant would be 3.5 cents/kWh.

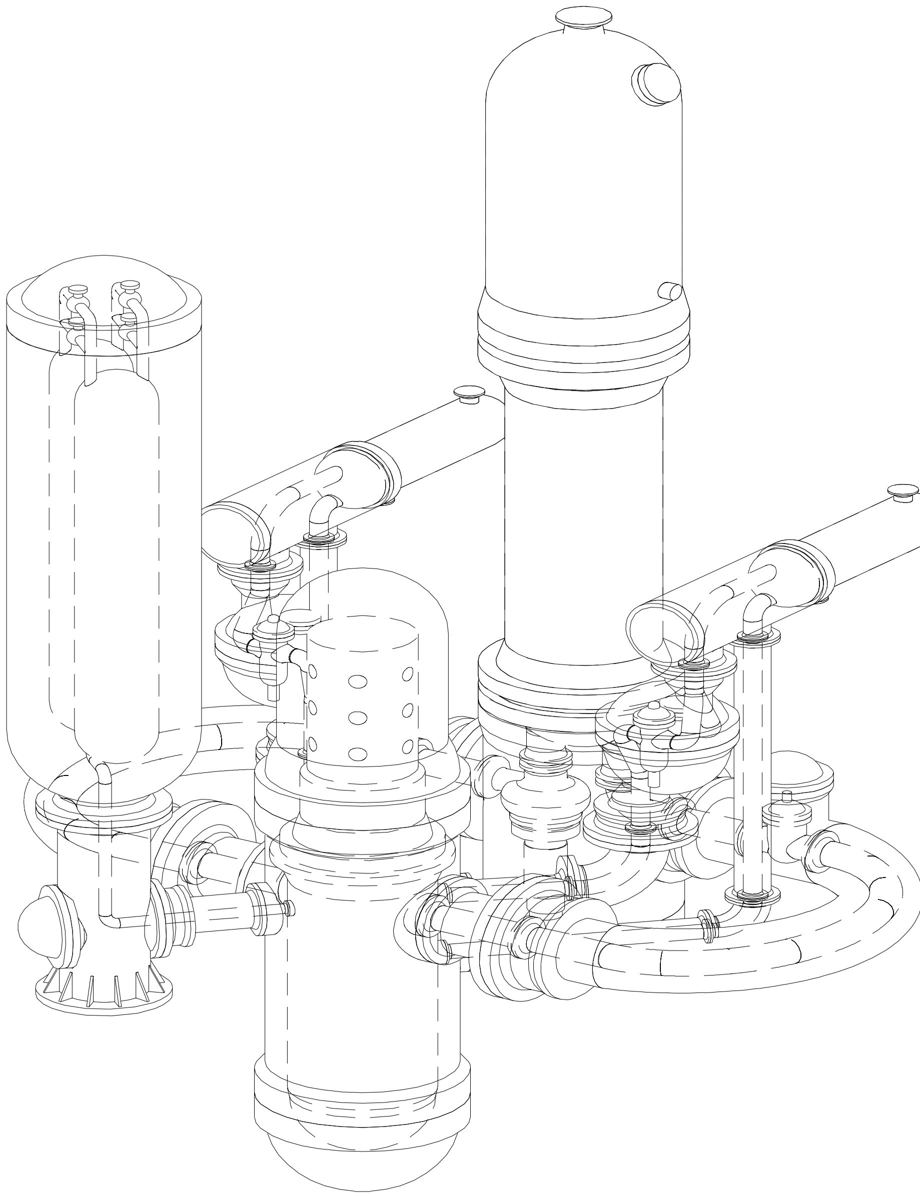

Figure 2. MARS Primary Cooling System (enveloped by the pressurized containment).

W2.3. POTENTIAL FOR CONCEPT MEETING GENERATION IV GOALS

In the following sections, the Loop PWR concept set is assessed against the Generation IV goals. The advantages and/or disadvantages of the Loop PWR concept set are evaluated relative to a typical Generation III reactor (e.g., the AP-600, ABWR, and System80+ designs), which serves as the reference system. In those areas for which no appreciable differences can be identified between the Loop PWR concept set and the reference, the analyzed concept is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and metrics by means of a label in parenthesis.

W2.3-a Evaluation Against High Level Criteria

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

The Loop PWRs exhibit the following advantages in the area of natural resource utilization:

- One of the concepts (Bloc Type PWR) adopts a long thoria-urania fuel cycle (10years). However, the thoria must be enriched with U-235 and that requires about 85% of the mining as for a conventional uranium fuel cycle. (SU1-1)

- Because MARS operates at a lower power density and thus lower fuel temperatures than current LWRs, it may be possible to achieve somewhat higher burnup levels resulting in more electric energy generated per unit mass of natural uranium. (SU1-1)

- If plutonium-based MOX fuels are utilized, it is possible to significantly increase the amount of electric energy generated per unit mass of natural uranium. (SU1-1)

Plutonium breeding is not possible in these thermal reactors and thus the utilization of natural uranium resources is limited compared with fast reactors. Also, the Bloc reactor's thoria-urania metal-matrix fuel cycle is not an intrinsic system characteristic. (In other words, the thoria-urania metal-matrix fuel being considered for the Bloc design could be used in an ALWR.) Also, the MARS concept does not propose the use of very long fuel cycles. Therefore, it is concluded that the Loop PWRs are essentially equivalent to the reference ALWRs in the area of Sustainability-1.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

The Loop PWRs exhibit the following advantages in the area of waste minimization:

- The Bloc Type PWR uses a long thoria-urania fuel cycle (10years). This significantly reduces the high level waste volume. (SU2-1)

- The Bloc Type PWR adopts a thoria-urania fuel cycle with a boron free core, and MARS utilizes low enriched uranium fuel with less boron in the core than an ALWR. Both approaches reduce boron waste. (SU2-1, SU2-2)

- Because MARS operates at a lower power density and thus lower fuel temperatures than current LWRs, it may be possible to achieve somewhat higher burnup levels resulting in less high level radioactive waste generated per unit mass of natural uranium. (SU2-1)

The Loop PWRs exhibit the following disadvantages in the area of waste minimization:

- The low thermal efficiency of the MARS system results in a larger amount of high-level waste per unit electric energy generated than the reference ALWRs. (SU2-1)

- For given installed capacity, the multi-module MARS plants are expected to have more activated materials (such as in-pile structures and instrumentation) than large monolithic plants. (SU2-1)

Because spent fuel is the radioactive waste of greatest concern, it is concluded that the Loop PWRs with uranium fuel are essentially equivalent to the reference ALWRs in the area of Sustainability-2. However, the use of long life thoria-urania cores will make the concepts better than the reference ALWRs.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

The Loop PWRs exhibit the following advantages in the area of proliferation resistance:

- MARS is based on the traditional LWR fuel cycle, which has proven to be proliferation resistant over the past four decades. The use of low-enriched uranium and the lack of reprocessing makes diversion of the Loop PWR fuel a relatively unattractive path to proliferation. (SU3-1).

- The U-233 generated in the Bloc thoria-urania fuel cycle is denatured with U-238 and protected by the U-232 that is generated, making it unusable as a weapons material. (SU3-3)

- The thoria-urania fuel cycle proposed for the Bloc concept produces relatively little plutonium and the long fuel cycle produces plutonium with very unattractive isotopics. (SU3-3)

The Loop PWRs exhibit the following disadvantages in the area of proliferation resistance:

- The Bloc thoria-urania in a metal matrix fuel requires UO_2 enriched with about 20% U-235. It is much easier to get to weapons grade material from 20% enriched UO_2 than from low (5%) UO_2 (24 versus 69 SWU per kilogram of 93% U-235), if enrichment facilities are available and misused. (See Appendix W8 for a further discussion of thoria-urania fuel.) (SU3-1)

It is concluded that the Loop PWRs are essentially equivalent to the reference ALWRs in the area of Sustainability-3. However, the use of long life thoria-urania cores will make the concepts better than the reference ALWRs.

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

This concept set has the following advantages in the area of safety and reliability:

- Because MARS operates at a lower power density than current LWRs, the margin to CHF at steady-state conditions and during transients is larger. (SR1-3)

- Lower temperatures in the MARS fuel may result in better fuel reliability as well as a lower release of fission gases upon fuel pin failure. (SR1-2, SR1-3)

- The use of a thoria-urania metal matrix fuel in Bloc will better retain fission products in the fuel and minimize the release of fission gases upon fuel pin failure. (SR1-2, SR1-3)

- Less or no boron core reduces the radioactive waste. (SR1-1)

This concept set has the following disadvantages in the area of plant reliability:

- Maintenance of the primary system might be more difficult than in the reference ALWRs because of encapsulation of the major components. (SR1-2)

- The presence of an additional vessel complicates the refueling operation, which is traditionally the main source of radiation exposure for the plant personnel. (SR1-1)

- For given power output, multi-modular MARS plant is likely to have more components (e.g. pumps, SGs, valves) than a large monolithic plant. (SR1-3)

It is concluded that since the essential innovative characteristic of these concepts is the use of a safeguard vessel to envelope the whole primary system, the reliability of these concepts might be worse than for the reference LWRs.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

Loop PWRs have the following advantages in the area of Core Damage Frequency

- The additional vessel filled with water eliminates the LOCAs as a safety concern. (SR2-3)

- The decay heat can be removed passively. (SR2-3)

- No boron (or less boron) used results a large negative moderator temperature reactivity feedback. (SR2-3)

It is expected that the core damage frequency of these systems will be much smaller than the reference ALWR because of the elimination of LOCAs, the passive decay heat removal, and the large negative moderator coefficient. It is concluded that Loop PWRs are better than the reference ALWR, with low uncertainty, in the area of Safety and Reliability-2.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

The Loop PWRs exhibit the following advantages regarding elimination of the need for offsite emergency response:

- In-vessel core catcher, double vessel, passive hydrogen control system (Bloc Type), encapsulation of the major components (Bloc and MARS), and a larger water inventory per unit power, all of which will result in better mitigation of severe accidents. (SR3-1)

It is concluded that Loop PWRs are better than the reference ALWRs, with low uncertainty, in the area of Safety and Reliability-3.

Economics–1. Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

Loop PWRs have the following advantages with respect to Economics-1:

- The loop PWRs have less active safety equipment, and therefore, lower maintenance costs for the safety systems. (EC-3)

- The loop PWRs use little or no boron. (EC-3)

Loop PWRs have the following disadvantages with respect to life-cycle cost:

- Maintenance of the major components could be a more difficult because of encapsulation of the system within an additional vessel. (EC-3)
- The MARS concept with its lower thermal efficiency will have a higher fuel cost per unit electric energy generated than the reference ALWRs. (EC-3)

The evaluators believe that at this point it is impossible to assess how the Loop PWRs will perform in terms of operating costs compared with the reference ALWRs.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

Loop PWRs have the following advantages with respect to capital costs:

- Low development costs. (EC-4)
- Elimination of some safety systems compared to the reference ALWRs. (EC-1)

Loop PWRs have the following disadvantages with respect to capital costs:

- Possibly higher capital costs than the reference ALWRs because of the encapsulation of the primary system. (EC-1)

The evaluators believe that at this point it is not possible to assess how the Loop PWRs will perform in terms of capital costs compared with the reference ALWRs, however, it is likely that the capital costs for a Loop PWR will be higher.

W2.3-b Summary of Concept Potential (Strengths & Weaknesses)

The potential of the Loop PWRs is summarized in the following Table 2:

Table 2. Summary of the concept strengths and weaknesses.

Category	Strength	Weakness
Sustainability	- Sustainability is good with thoria-urania once-through fuel cycle. - Use no boron or less boron.	- MARS has a low thermal efficiency.
Safety & Reliability	- Low core damage frequency. - Passive safety systems. - Good mitigation of severe accidents	- Some difficulties in maintenance due to compactness and encapsulation.
Economics	- Potential for high capacity factor - Simplicity reduces overall volume. - Potential for low capital cost ($/kWe) for the Bloc PWR (>1500 MWe).	- MARS has a low thermal efficiency. - Cost of the additional vessel - Uncertainty in financial risk. - Some difficulties in maintenance due to compactness and encapsulation.

W2.4. TECHNICAL UNCERTAINTIES

W2.4-a. Research and Development Needs

In general the small Loop PWR (MARS) does not need much R&D, but needs some engineering to deploy the concept. The large Loop PWR (Bloc Type) needs considerable R&D efforts and this concept could be deployed by 2030. A list of the technical uncertainties and R&D needs that, if addressed, would benefit these concepts follows:

1. Maintenance free components
2. Thoria-urania dispersed zirconium metal fuel and clad material
3. Boron free operation
4. Double vessel and bloc type design and seismic responses
5. In-vessel hydraulic control rod drive mechanism
6. Various passive safety systems in large PWR
7. Nano particles in coolant
8. In-vessel retention
9. Fully automated I&C system
10. Evaluation of economic viability.

W2.4-b. Institutional Issues—Licensability & Public Acceptance

Public acceptance issues for the Loop PWRs have not been identified. However, the public should be receptive to the elimination of LOCA, the mitigation of severe accidents, the passive safety performance, and the possibility of eliminating the need for the emergency response. In addition, by utilizing Thorium based fuel it is possible to reduce the production of plutonium and make better use of the natural resources.

The licensability of this concept set is facilitated by the use of existing PWR technologies, where applicable. For those concepts, systems, and components that are new, it will be necessary to conduct supporting experiments/tests to demonstrate, verify, and validate their performance and reliability.

W2.4-c. Timeline for Deployment

The proponents of the MARS reactor claim their system is almost ready for deployment (within 10 years). On the other hand, Bloc PWR with thorium fuel needs considerable R&D activities, but with adequate funding it could be deployed by 2030.

W2.5. STATEMENT OF OVERALL CONCEPT POTENTIAL

This reactor concept set offers potential for superior safety compared with the reference LWRs. However, issues to be resolved include the reliability and maintenance of the primary system components that are not easily accessible, and the impact of the additional vessel on the capital costs.

The concept is retained for further assessment.

W2.6. REFERENCES

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W2.7. TOP-TIER SCREENING SHEET—LOOP PWRS

Summary Evaluation: _X__ Retain _____ Reject

Goal		--	-	+	++	Comments
SU1	Fuel Utilization		E			- Equivalent to reference ALWR when an all-uranium fuel is used.
SU2	Nuclear Waste		E			- Equivalent to reference ALWR when an all-uranium fuel is used.
SU3	Proliferation Resistance		E			- Equivalent to reference ALWR when an all-uranium fuel is used.
SR1	Worker Safety and Reliability					- Reliability and maintenance of primary system components is questionable - More complicated refueling may result in higher worker exposure
SR2	CDF					- Elimination of LOCAs, passive decay heat removal, and large negative moderator coefficient
SR3	Mitigation					- In-vessel core catcher, double water filled vessels, and larger water inventory
E1	Life-cycle cost					- High plant capacity factors, less wastes, but the potential for higher fuel and O&M costs
E2	Capital Cost and Financial Risk					- Cost of the additional vessel - Simplification of the nuclear island - Low development costs

Appendix W3

Simplified Boiling Water Reactors Concept Set Report

December 2002

Appendix W3: Simplified Boiling Water Reactors

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ABSTRACT

The attributes of the simplified boiling water reactor (SBWR) group, a subset of the water-cooled concepts submitted to the Generation IV Project Technical Working Group 2, are summarized. In addition, their potential for meeting the Generation IV Roadmap goals are discussed, their technical uncertainties are addressed, and an initial assessment of their research and development needs is given.

This group includes classical direct cycle boiling water reactor (BWR) concepts, simplified in design compared to today's commercial BWRs. The identified candidate concepts within this group are all founded on existing and proven BWR technology, but with design improvements and advanced features intended to provide economic or other advantages. They cover a very wide range of plant power ratings (50 to 1380 MWe). Passive safety features are used extensively in these design concepts.

W3.1. INTRODUCTION

W3.1-a. Background and Motivation for the Concept

Potential Generation IV boiling water reactors (BWRs) are discussed. BWRs are a major component of the existing commercial fleet in most industrialized nations. Steady refinement and advancements to the existing plant designs have culminated in two BWR-type advanced light water reactor (ALWR) designs. One of these is the Advanced BWR (ABWR), an ALWR of the "evolutionary" category, design certified in the U.S. by NRC, with units constructed and operating successfully in Japan and under construction in Taiwan. The other is the SBWR, a simplified, passively safe ALWR, developed conceptually as part of the US ALWR Program.

Proven technology is the firmest foundation for development of future, improved reactor technologies. The success of the ABWR concept (excellent safety and above average capacity factors) is itself the strongest argument for continuing to develop and deploy state-of-the-art BWRs. The BWR design candidates in this group include key features of the existing successful BWRs, the advanced BWR concepts already developed, and other innovative features. The candidates fall into three categories: modular (or quasi-modular), monolithic, and special-purpose. Although the candidates will be summarized individually, conclusions will be formulated on the SBWRs as a group.

W3.1-b. National and International Interest

The BWR designs, successfully promoted by the General Electric Co and their licensees, have been popular from almost the beginning of the commercial nuclear era. The Generation II concepts, perhaps best represented by the BWR-6, have been eclipsed by the more technically advanced ABWR design—a Generation III plant. Because of the established record of success achieved by the BWR designs, there is every reason to believe that there will be commercially-successful Generation IV SBWR designs. The designs submitted for consideration are summarized in Table 1. Of the 5 designs, there is one monolithic design submitted by GE, three modular designs (2 from the US and 1 from Japan), and one special purpose concept designed to desalinate water (from Japan).

The best known of the submitted concepts are the European Simplified BWR (ESBWR), submitted by GE (W13), and the SBWR design submitted by Purdue University (W8)—since Purdue's design is based substantially on the original GE SBWR design that was submitted as a licensing candidate a few years ago. The U.S. Nuclear Regulatory Commission did not grant a license to the GE's SBWR design since GE withdrew it from consideration before the process was very far advanced.

Organizations in Europe, the US, Japan, and Taiwan have expressed interest in the SBWR design.

W3.2. CONCEPT DESCRIPTION

The candidate BWR design concepts within this group are:

1. SMART (Concept W7)
2. SBWR-Purdue (Concept W8)
3. LSBWR (Concept W23)
4. ESBWR (Concept W13)
5. Desalination Plant (Concept W22).

Table 1. Summary of simplified boiling water reactor concepts submitted to DOE for the Generation-IV Program.

Gen-IV Designation	Proposer	Size	Coolant State / Pressure	Containment	References
W7 (SMART)	Khatib-Rahbar (Energy Research, Inc, USA)	50-300 MWe	Boiling,	Large volume BWR/PWR hybrid	Khatib-Rahbar 2001.
W8 (SBWR-Purdue)	Ishii (Purdue University, USA)	50 MWe	Boiling; 7.2 MPa	Small	Ishii, et al. 2001a; Ishii, et al. 2001b.
W23 (LSBWR)	Heki (Toshiba, Japan)	300 MWe	Boiling; 7.0 MPa	Smaller than conventional BWR (with suppression pool)	Heki, et al. 2001.
W13 (ESBWR)	Rao (General Electric, USA)	1380 MWe	Boiling	Large (with suppression pool)	Rao 2001.
W22 (Desalination)	Kataoka (Toshiba, Japan)	589 MWth	Boiling; 7.0 MPa	Small (with suppression pool)	Kataoka 2001.

Significant common features of the group are as follows:

- These BWRs are all direct cycle light water reactors with conventional energy conversion systems and efficiencies (with the exception of the desalination plant, W22).
- All rely on natural circulation, rather than on mechanical or jet pumps, either internal or in re-circulation loops.
- All utilize passive safety features similar to those used in the reference plant (ABWR).
- All but one of the concepts use relatively conventional uranium oxide, Zircaloy clad fuel. The SBWR-Purdue, Concept W8, expressed a preference for 5% enriched ThO2-UO2 fuel. However the backup fuel for this concept is low-enrichment uranium (LEU).
- The remaining SBWR power reactors, although specifying LEU as their chosen fuel, do mention backup fuels that are: ThO2-UO2 (SMART), medium-enriched UO2 for very high burnup (LSBWR), and MOX rods (ESBWR).
- All the modular concepts feature long fuel cycles ranging from 10 years (SBWR and SMART, W8 and W7) to over 15 years (LSBWR, W23). Due to its 15-year fuel cycle, the LSBWR design does not include a spent fuel pool. The ESBWR concept (W13) features intermediate length fuel cycles. Refueling must be accomplished with the system offline.
- The modular concepts are designed, to one degree or another, to complete a major portion of the system construction in a factory. The factory-produced system is then transported and deployed at

the site. Examples of this approach are SMART (W7) and SBWR (W8). Although not clear in the concept description, portions of the LSBWR (Concept W23) seem to be factory constructed.

- All the SBWR concept designs have bottom-entry control rod drives with the exception of SMART (W7) and LSBWR (W23) which has an internal top entry design.

- The containments fall into 2 general categories: large volume—BWR/PWR hybrid (SMART, W7) and volumes of various sizes with suppression pools (W8, W13, W22, and W23).

The concepts differ in size and structural approach, covering both modular and monolithic designs of power ratings from 50 to 1380 MWe. They also differ significantly in safety system design, in plant layout and equipment configurations, in containment design, in operating characteristics, and in level of design maturity (some are highly conceptual, while others are well developed). The predominant features of these five concepts are listed and compared in Table 1 above.

In the following summaries, the Simplified BWRs have been grouped into three categories: modular (concepts W7, W8, and W23), monolithic (W13), and special-purpose (W22). The developers of the concepts primarily wrote the concept summaries reported below. They have been edited for style and brevity. Some of their statements may not reflect the judgment of the Technical Working Group, which is reported instead in Section 3 of this appendix.

W3.2-a. Modular SBWRs (W7, W8, & W23)

Modular SBWRs are small- or medium-size BWRs (50-300MWe) designed to have major components manufactured in factories and then shipped in toto to the plant site. The degree to which each of these concepts will be completed in a factory and then shipped to the plant site differs from one to another—and was not well-defined in the concept descriptions. The modular BWRs, as a group, increase proliferation resistance by tending to have long operating cycles.

A brief description of the concepts, complied from information supplied by the authors, follows.

W7—SMART

SMART (**S**mall **M**odular **A**dvanced **R**eactor **T**echnology) is a Boiling Water Reactor (BWR) that is being designed by Energy Research, Inc. to include the following characteristics:

1. Low-power and high-efficiency over a range size

2. Fuel and core designs with long operating cycles

3. Passive, built-in safety and environmental systems

4. Scaleable (in the range of 50 to 300 MW(e)), compact, reliable, and safe design

5. Easily transportable and deployable at the site.

The SMART concept consists of a BWR system with a large volume containment that is more typical of pressurized water reactors. The BWR system concept reduces the number of system components and complexity (no secondary steam production system). When the economic study is completed, it is anticipated that the elimination of the secondary system will more than offset the cost that stems from the use of a stronger containment.

The core design allows the use of either low-enrichment uranium or a mixture of low-enrichment uranium and thorium. Both fuels are relatively proliferation-resistant, and in conjunction with advanced fuel pin and core materials, the design should allow continued operation (with provisions for on-line maintenance) for periods exceeding 10 years, without refueling. The emergency core cooling system (ECCS) and the containment heat removal system are based on passive natural circulation: water for emergency core cooling, and air flow (aided by evaporative cooling at powers higher than 100 MWe) for containment heat removal.

The vessel and containment systems are shown in Figure 1. The feed-water system is designed to achieve the desired re-circulation, core cooling and power production without the need for internal jet and external re-circulation pumps. The system is equipped with a Core Automatic Depressurization System (CADS). Reactor pressure control will be accomplished by means of relief valves which discharge through spargers submerged in a large In-Containment Water Pool (ICWP). The same discharge lines are used for automatic depressurization of the vessel during accidents. The borated water contents of the ICWP will also be used to reflood the vessel (by gravity) once it has depressurized, as well as to flood the reactor cavity/pedestal region (for vessel lower head cooling), in case of severe accidents. The steel containment will serve as the ultimate heat sink, which will also utilize a passive cooling system based on natural circulation of air on the exterior of the steel containment shell. At power levels exceeding 100 MWe, the design uses a combination of air and evaporative cooling (<u>not</u> involving any water sprays on the containment shell and not shown in Figure 1). Condensate on the shell's interior is returned to ICWP, where it is available again for vessel or cavity flooding. Other design features include: (a) the use of passive engineered design features that are intended to deal with severe accidents as part of the design basis envelope and (b) risk-optimization (eliminating the deficiencies in the current generation designs, through simplifications and innovation).

Design feasibility studies performed at Energy Research, Inc. have demonstrated the:

- Overall feasibility of the concept
- Achievement of very long fuel operating cycles (more than 10 years for uranium and more than 6 years for thorium fuels)
- Relatively slow progression of accidents/events
- Effectiveness of the various engineered systems (ECCS and the passive containment heat removal system)
- Effectiveness of the various engineered systems to deal with severe accidents
- Slow (typically days to weeks) pressurization potential of the reactor containment—such that it always remains below the containment design pressure
- Adequacy of the existing technology (the fuel design requires additional R&D to achieve very high burn-ups) to support the design certification, fabrication and ultimate construction.

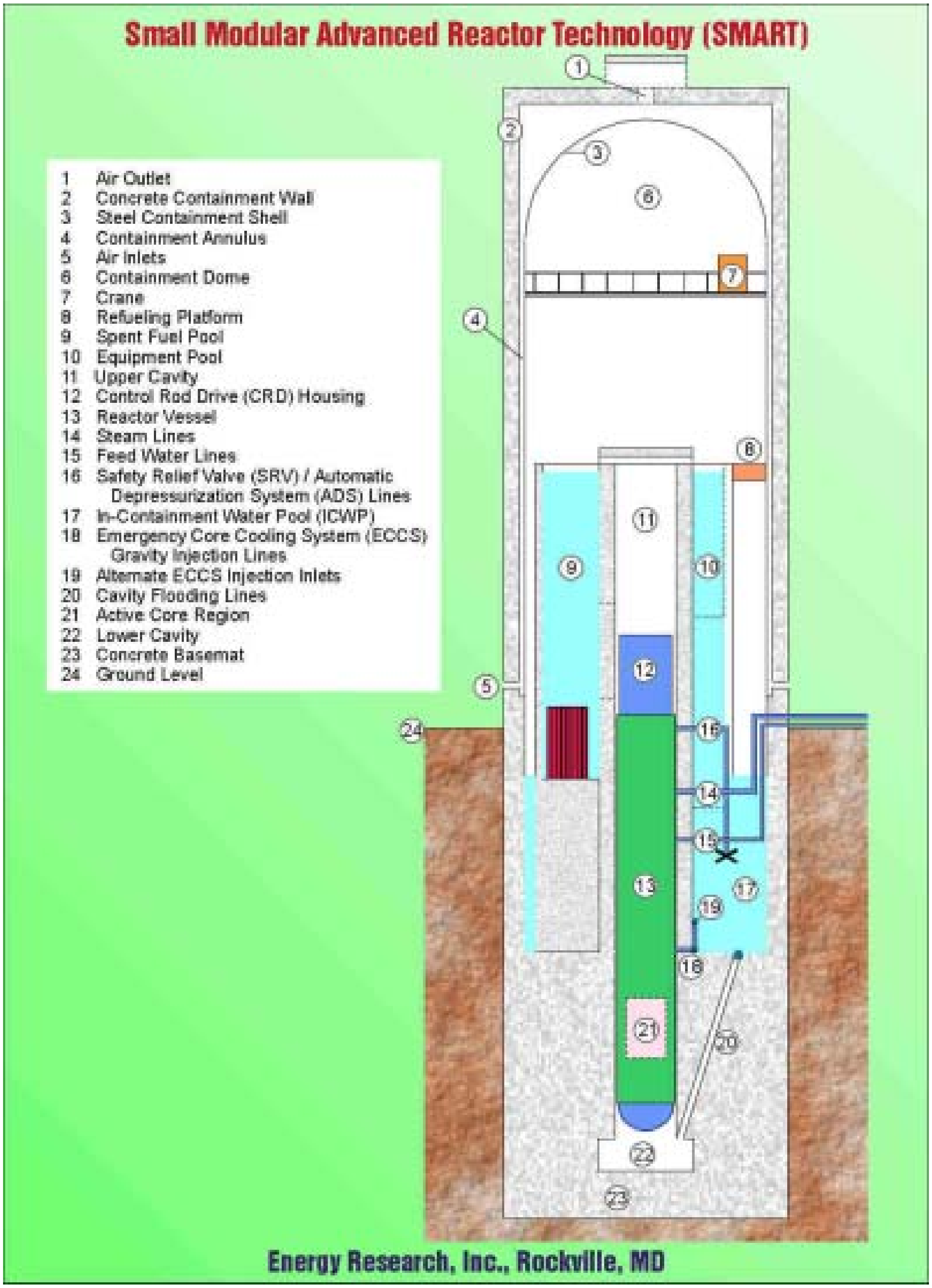


Figure 1. Schematic of the SMART primary system, safeguard vessel, and emergency systems.

Additional benefits that could be realized, include provisions in the design to build several reactor modules at the lower-end of the power range, within an integrated nuclear plant infrastructure, in a common (but larger) containment with associated decay heat removal and other systems. This would enable demand-based expansion of a reactor plant site equipped with many SMART modules, over time.

W8—SBWR-Purdue

The Simplified Boiling Water Reactor (SBWR) was initially developed by GE and supported by DOE. The original SBWR design incorporated advances in existing proven technologies that have been developed over many years of commercial nuclear plant operation. Researchers at Purdue have continued to improve this design by adding new features that support the Generation IV reactor goals.

The most important feature of the SBWR-Purdue is the elimination of re-circulation loops and pumps. The core is cooled by natural circulation cooling which results in an extremely reliable and simple system for steam production. The engineered safety systems are mostly passive, there are no active emergency core cooling systems (ECCS). The ECCSs are based on gravity-induced flow. Furthermore, the containment cooling is also achieved with passive systems. Elimination of the re-circulation pumps and loops, internal pumps, and active safety systems substantially reduces the number of piping and valve components. Furthermore, the design eliminates the need for a large emergency AC power supply as well. This simplification has considerable potential for reducing the cost for the reactor. In addition, the passive safety systems are much more reliable than the active systems and they provide enhanced safety against loss of coolant accidents and other design basis accidents.

Because the SBWR-Purdue is a passively cooled reactor, it has a number of advantages. First, the SBWR uses a direct Rankine cycle, which eliminates the need for steam generators. Second, a significant reduction in the number of pumps and the elimination of the requirement for an emergency AC power supply simplifies the plant design, operation and maintenance, as well as the overall cost. However, because the net power production is low (~50 MWe), the economics of the design are subject to considerable uncertainty.

The reactor safety systems in the SBWR-Purdue are shown in Figure 2 and consist of the automatic depressurization system, the gravity driven cooling system, drywell, suppression pool, the containment cooling system, and the isolation condensers. The automatic depressurization system is designed to rapidly depressurize the vessel following the receipt of a low vessel water level signal. This system is made up of both safety relief and depressurization valves. The depressurization of the reactor vessel allows gravity injection from the gravity driven cooling system. At the same time heat is removed by the flashing of coolant in the reactor vessel. For long term cooling of the drywell, several condensers have been adopted as a passive containment cooling system. The steam from the drywell is condensed in the passive containment cooling system condenser and is returned to the reactor vessel. The passive containment cooling system non-condensable vent line purges non-condensable gas into the suppression pool.

Purdue University has made several new design improvements to the current SBWR technology. Specifically three modifications have been made to the SBWR design to address the Generation IV goals. First, a $(Th+U)O_2$ cycle has been adopted to address the nonproliferation requirements. This mixed fuel has several advantages. The presence of the thorium reduces the build up of plutonium in the fuel. Also, at the end of the burn up period the spent fuel will contain U-232, which decays to highly radioactive products and helps make the fuel proliferation resistant. Also, by reducing the power density, the fuel cycle length can be extended up to ten years. A preliminary design of the Purdue 50MWe SBWR is described by Ishi et al. (2001a). Calculations for the 50 MWe modular SBWR (Table 2 in Ishii et al 2001b) show increased fuel life cycle with substantial improvement on negative void coefficient.

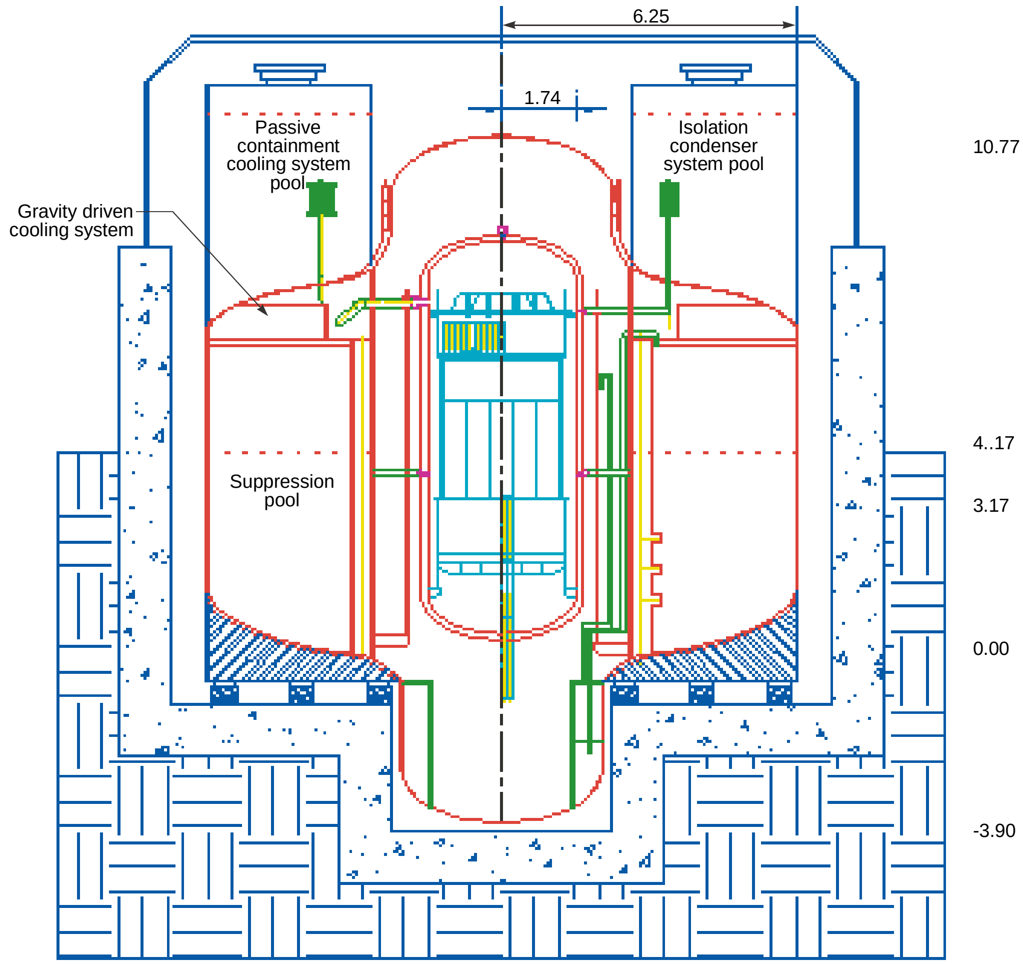


Figure 2. The Purdue 50MWe modular SBWR design, heights are shown with respect to the reactor pressure vessel bottom. All dimensions are in meters.

Secondly the original vacuum breaker check valve design has been changed to a passive system consisting of a discharge pipe submerged in a separate water pool, which is connected between the suppression pool and drywell. The water head for gas flow from drywell to suppression is very high and hence direct venting from drywell to the suppression pool does not occur thorough this vent line. The submergence of the discharge pipe determines the pressure difference required to vent the gas from the suppression chamber to drywell. The passive design improves safety and reliability. In the original GE design, the vacuum breaker is a mechanical check valve. This check valve has a potential for malfunction that may lead to ineffective cooling of the containment.

The third improvement is a reduced-size passive refill system for the isolation condenser pools. The isolation condenser pools sit above the containment and house the isolation condensers and passive containment cooling system condenser units. The isolation condenser pool volumes were reduced to prevent enlargement of the containment building and thereby decrease the capital costs. To maintain an adequate cooling water supply in the smaller isolation condenser pools, the SBWR-Purdue design contains a passive refill system that functions as follows. During a hypothesized accident with containment pressurization, the isolation condensers and/or passive containment cooling system condenser units condense steam from the drywell and reactor pressure vessel. The condensers are cooled with boiling water at atmospheric pressure. The drywell pressure is typically 200–250 kPa following a

LOCA. The pressure difference between the condenser pressure and the containment pressure drives the steam through the heat removal units. The isolation condenser pool water boils off and the water level decreases. The new passive refill feature takes advantage of the pressure drop in the inlet steam line to rotate a small turbine blade that has been added to the steam line. The turbine's rotating shaft extends outside the containment and powers a pump that replenishes the isolation condenser pools with water from other sources.

W23—LSBWR1: Long Operating Cycle SBWR

A long operating cycle simplified BWR (LSBWR) is being developed by Toshiba Corporation and the Tokyo Institute of Technology. Major characteristics of the LSBWR are:

- No refueling or shuffling (target : over 15 years), resulting in:

  High availability

  Elimination of fuel pool and refueling machine

  Ease of operation.

- Natural circulation BWR with bottom located core, internal control rod drives from the top, and a passive containment vessel with a passive cooling system, resulting in:

  An in vessel retention capability

  A large water inventory above core region

  No passive containment vessel vent thereby providing a high degree of inherent safety.

- Reactor and turbine systems in one common building, resulting in:

  A highly modular arrangement in the hull structure (ship frame structure)

  Ease of seismic isolation

  Standardization and factory fabrication.

The LSBWR design aims to achieve economical competitiveness using the above features. The LSBWR design concept is shown in Figure 3.

The long operating cycle (over 15 years) is achieved using a high conversion core via a combination of medium enriched uranium oxide fuels and non-tight lattice bundle since this configuration encourages natural circulation for core cooling. Thus, the core has the following characteristics: (a) an extension of reactivity life using fixed type burnable poison, (b) and increase of control rod worth using smaller fuel bundle sizes, and (c) an extension of the control rod life using increased neutron absorber material in the control rods.

Other important features include:

- No evacuation requirements during severe accidents since highly reliable equipment and systems are used such as:

  - A large reactor pressure vessel inventory

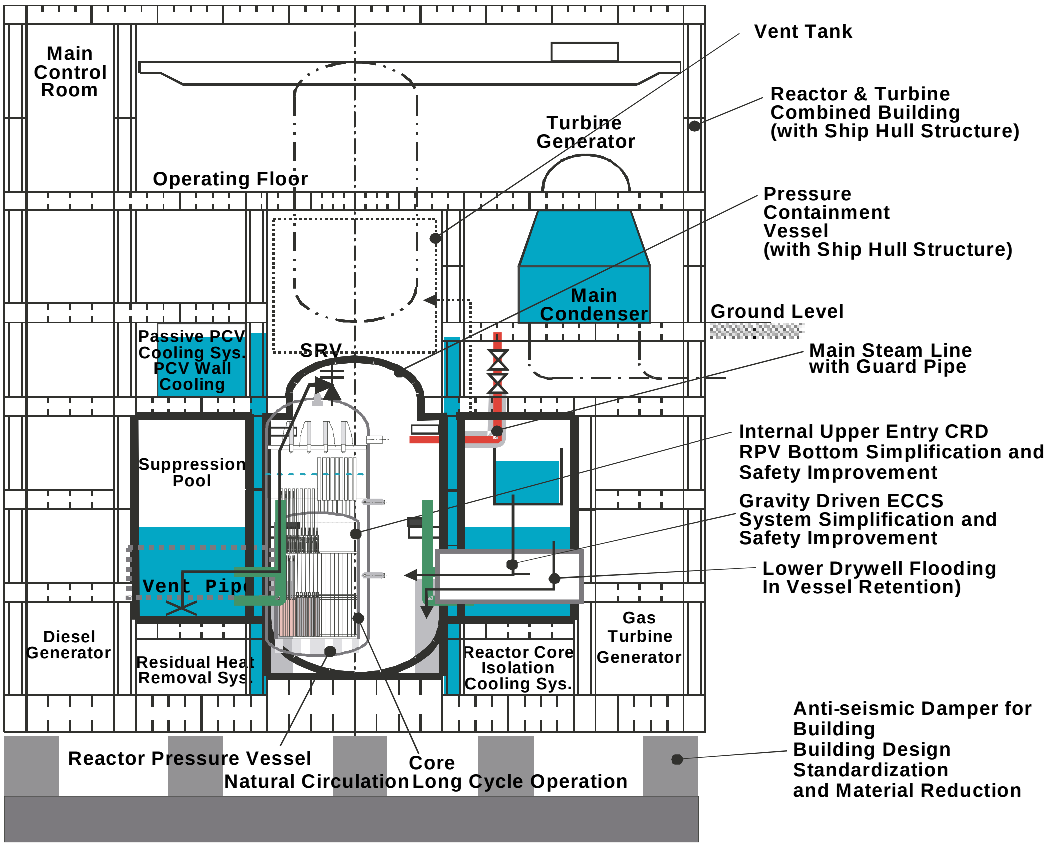


Figure 3. Long operating cycle SBWR concept.

- Bottom core configuration
- In-vessel retention capability
- A passive emergency core cooling system and passive containment vessel.

- Natural circulation core cooling**:**

  The lack of re-circulation pumps simplify the design and result in better operational reliability.

  A simplified steam separator improves the natural circulation driving force (option).

- Internal upper entry control rod drives:
  - The bottom located core results in a large water inventory above the core for an increased natural circulation potential and a large safety margin in the event of the loss of inventory.
  - The reactor pressure vessel and passive containment vessel heights are shortened.
  - Penetration of the control rod drives through the reactor pressure vessel top or bottom head is not required.

- Passive containment vessel configuration:

  - Safety relief valves are placed on the reactor pressure vessel head to enable the drywell diameter to be minimized.

  - The drywell air space is minimized and contains only safety-relief valves and depressurization valve components, the gravity driven core cooling system, and the lower drywell flooding piping.

  - The main steam and feed-water piping is routed through the suppression pool air spaces. The pipes are protected by guard pipes.

  - The gravity driven cooling system piping is contained in the access tunnel placed in the lower part of suppression pool.

  - Isolation valves are installed outside the passive containment vessel.

- Gravity driven core cooling system: Since the reactor core is placed at the reactor pressure vessel bottom, the emergency coolant injection system, consisting of the depressurization valve and gravity driven core cooling system combination, will be highly reliable and water coverage of the reactor core following an accident will be assured.

- Lower-drywell flooding: Since the control rod drive housing tubes are removed from the reactor pressure vessel bottom, the outer wall of the reactor pressure vessel bottom can be easily cooled by flooding the lower-drywell with the suppression pool water in case of a severe accident. The molten core can be cooled and maintained in the reactor pressure vessel bottom by cooling the reactor pressure vessel wall.

- Drywell wall natural circulation cooling (use of ship hull structure passive containment vessel): The containment wall outer space and the ship hull structure is filled with cooling water which is boiled off to the atmosphere to passively cool the containment vessel during an accident. The containment wall cooling system is also used for the drywell cooling during normal operation and therefore the drywell arrangement is simplified and does not have the drywell cooling components used in the current BWR containments. When the cooling water in the passive containment cooling system pool above the containment vessel is exhausted, external pool water or seawater is supplied by gravitational force in this ship hull structure passive containment vessel wall space. Consequently, highly reliable and long term passive containment vessel cooling is achieved.

- Double cylindrical raised suppression pool: The double cylindrical raised suppression pool with the ship hull structure is installed around the cylindrical drywell and above the core elevation. This results in a simpler and stronger structure, and the suppression pool water can be easily used for gravity driven core cooling and lower drywell flooding.

- Vent tank: Because the need for a spent fuel pool is eliminated because of the super-long operating cycle, a vent tank is located above the passive containment vessel and is used as the suppression pool air space. Thus, a containment vent to the environment is not required to handle a severe accident. In addition, the passive containment vessel pressure can be decreased by depleting any flammable gas using a flammable gas control system.

- Design basis accident countermeasures: A passive containment vessel spray cooling system, using an active residual heat removal system and active single gas turbine set is used in addition to the gravity driven cooling system and the passive containment vessel wall cooling mentioned above. Using this equipment the passive containment vessel pressure following an accident can be quickly decreased to near atmospheric pressure to minimize the radioactive releases to the environment. Two residual heat removal system trains are enough for this configuration with the single failure assumption, and two small diesel generators (or gas turbine generators) are installed.

- Combined building concept with ship hull structure: This concept does not use a conventional steel concrete structure building—instead it uses a building fabricated using ship hull building techniques. Because the ship hull-type containment building has anti-seismic dampers, it can be standardized and thus the construction period can be shortened. The ship hull structure consists of steel plate with girders (large beam) and stiffeners (small beam). This design facilitates factory fabrication of a LSBWR module. As a result, the site work and construction period are reduced, and the production quality is improved. Thus, module ship hull structure and factory fabrication reduces the construction cost. In the LSBWR building design, the reactor building and the turbine building are combined into one building. Neither a fuel pool or fuel-handling machine is needed since the system has a long cycle operation, therefore, it is possible to mount turbine system on the upper part of the reactor building. A one building arrangement reduces the building volume and anti-seismic structures. This unique building concept results in an overall capital reduction.

W3.2-b Monolithic SBWRs (W13)

The ESBWR is a 4000 MWt (approximately 1400 MWe), boiling water reactor, submitted by the General Electric Co., that uses the same basic passive technology and simplified design as its predecessor (the 2000 MWt SBWR). The system makes use of existing technology when ever possible—such as GE’s fine motion control rod drive system. The ESBWR plant design relies on the use of natural circulation and passive safety features to enhance the plant performance and simplify the design (such as reductions in the required numbers of control blades and control rod drives). The use of natural circulation has allowed the elimination of several systems—such as the re-circulation pumps. Adequate natural circulation behavior has been achieved using shorter fuel and an improved steam separator (to reduce the core pressure drop), and a seven-meter chimney to enhance the driving head.

The ESBWR uses isolation condensers for high-pressure inventory control and decay heat removal under isolated conditions. The isolation condenser system has four independent high-pressure loops, each containing a heat exchanger that condenses steam on the tube side. The tubes are in a large pool, outside the containment. The steam line connected to the vessel is normally open and the condensate return line is normally closed. Refer to Figure 4.

In the event of an accident, the vessel is depressurized rapidly to allow multiple sources of safety and non-safety systems to provide water makeup. By eliminating all large penetrations in the lower part of the reactor vessel, the ESBWR core will remain covered by water during any rapid depressurization event. Hence, the makeup system has only to provide a slow water makeup to account for loss of inventory resulting from boil-off by decay heat. The makeup water flows into the vessel by gravity, using the Gravity Driven Cooling System, instead of relying on pumps and their associated support systems. The ESBWR uses an automatic depressurization system to depressurize the vessel.

Containment heat removal is provided by the Passive Containment Cooling System, consisting of four safety related low-pressure loops. Each loop consists of a heat exchanger open to the containment, a condensate drain line, and a vent discharge line submerged in the suppression pool. The four heat exchangers, similar in design to the isolation condensers, are located in cooling pools external to the containment.

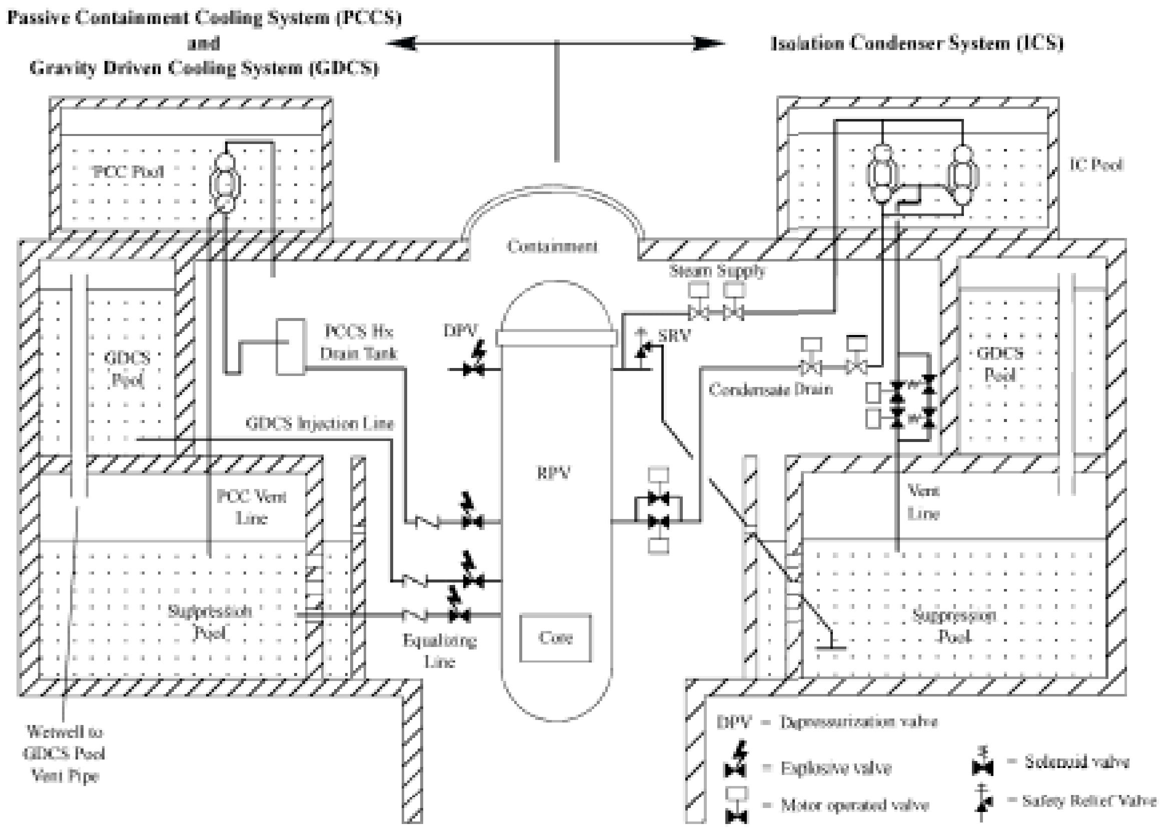


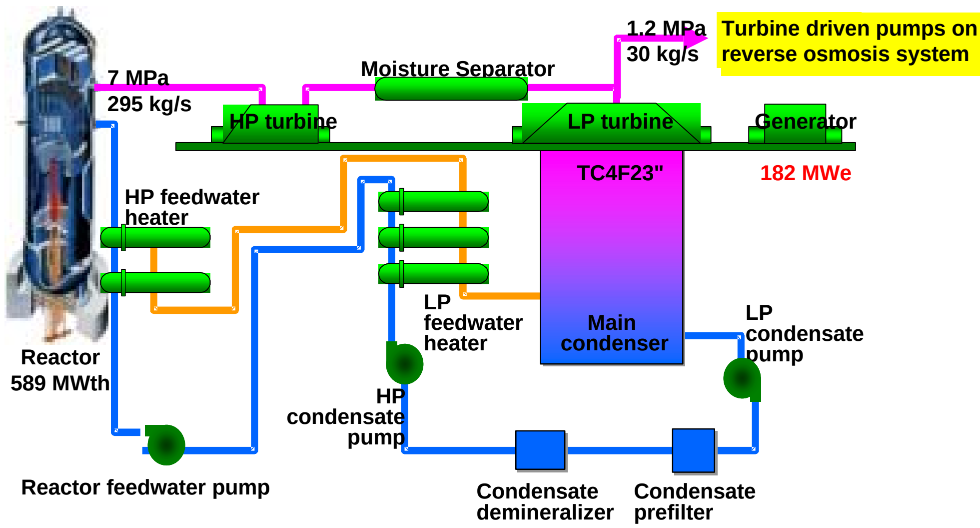
Figure 4. Schematic of Passive Safety Systems for the ESBWR.

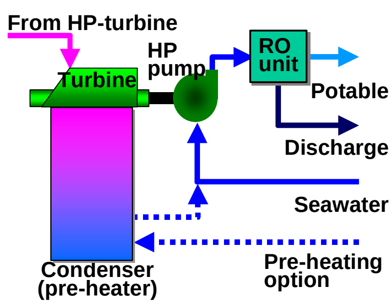
One key new feature effectively allows a larger wet well-to-dry well volume ratio, without significantly enlarging the containment. The gravity driven cooling system pools are located, topologically in the wet well, and therefore are sealed off from the dry well. Figure 4 shows a schematic of this design. The airspaces in the gravity driven cooling system pool region and the wet well are connected by pressure equalization lines. As a result of this connection, the additional airspace volume created by the gravity driven cooling system pool draining, is now available for wet well gas expansion. This keeps the containment pressure low following an accident.

The ESBWR is designed to deliver 1380 MWe using 1132, equipped with 2.7 long fuel. The total vessel height is 27.7 m (vs 21.1 m for the ABWR) and the vessel diameter is 7.1 m. The power density is 53 kw/l.

W3.2-c. Special Purpose SBWRs: Desalination (W22)

Concept W22 is a coupling between a small natural circulation BWR and a reverse osmosis seawater desalination system through turbine-driven-pumps as an interface (see Figure 5). Both the BWR and the reverse osmosis system are simple designs, that improve the economics as well as the plant reliability. The use of turbine-driven -pumps, which are often used in nuclear power plants, also enhances the economics as well as the safety because they can eliminate the use of an extra heat exchanger as an interface between the nuclear system and the desalination system. All these technologies are well proven and existing so that neither large R&D nor new investments in manufacturing facilities is necessary.





Reverse osmosis system

General	
Rated thermal/electrical power	589/182 MW
Water production rate	102×10^{3} m^{3}/d
Steam cycle	
Turbine type	TC4F23"
Stages of feed-water heating	HP×2, LP×3
Desalination	
Process	Reverse osmosis
HP pumps	TD×2 (50 %×2)
Capacity	17 MW/unit
Backup	MD×1 (50 %)

Figure 5. Schematic view of TTBWR and reverse osmosis plant

The standard BWR design is further simplified for use in a co-generation plant producing both electricity and potable water under the design principle: maximum utilization of proven technologies. The core power density is decreased instead of changing the core and/or fuel designs. This decrease in power density results in simplification in the coolant circulation system of the BWR because the natural circulation is high enough for a core with such low power density. The external re-circulation loops including re-circulation pumps are therefore not needed. The low power density also lengthens the refueling intervals and consequently enhances the availability of the plant. For example, a 48 effective-full-power-month cycle length is achievable with the standard 45 GWd/t BWR fuel*. Theoretically, the availability could exceed 95 % with a low power density core.

The safety systems developed for the most recent BWRs are further sophisticated to fit for this small natural circulation BWR with low power density. The emergency core cooling system configuration is sized to take advantage of the relatively small power of the core. The emergency power sources are diversified into two types: diesel generator or gas turbine generator, owing to relatively small capacity required for them. A passive containment cooling system is adopted for overpressure protection of the primary containment vessel in case of a severe accident.

The balance-of-plant consists of a turbine system generating 182 MWe and a seawater desalination system producing about 100×10^3 m^3/d of potable water as a reference design. The turbine system uses a regenerative steam cycle consisting of two stages of high-pressure feed-water heating and three stages of low-pressure feed-water heating. A portion of the steam (30 kg/s) is bled after the high-pressure turbines and used to drive two turbine-driven-pumps. All the systems are designed based on existing technologies.

Instead of backup boilers, which are often used in distillation seawater desalination systems, a motor-driven pump is used for backup. Because the motor-driven-pump is powered by external sources, backup boilers together with the associated systems are unnecessary.

This reverse osmosis system, including the turbine-driven-pump interface, has advantages in efficiency, economics, and safety over conventional distillation systems for seawater desalination. This reverse osmosis system produces about 100×10^3 m^3/d of potable water while a distillation system would produce only about 80×10^3 m^3/d if the same amount of steam is used. Only a motor-driven -pump is added for backup of the reverse osmosis seawater desalination while backup boilers, together with their associated systems including fuel tanks, are necessary for a distillation system. Because the possibility of radioactive contamination of the seawater from the BWR steam is physically eliminated, no extra barrier is necessary for this reverse osmosis system. A distillation system would have an extra barrier (extra heat exchanger) to decrease the failure probability and to mitigate problems stemming from the thin-wall heat exchanger tubes that separate the BWR steam and seawater. Therefore, the reverse osmosis system is a better nuclear seawater desalination system than a classical nuclear distillation system.

W3.3. POTENTIAL FOR CONCEPT MEETING GENERATION IV GOALS

In the following sections, the SBWR concept set is assessed against the Generation-IV goals. The advantages and/or disadvantages of the SBWR concept set are evaluated relative to a typical Generation-III reactor (in this case, we are primarily comparing the SBWR with the ABWR). In those areas for which no appreciable differences can be identified between the SBWR concept set and the reference, the analyzed concept is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and metrics by means of a label in parenthesis.

* Extended operation of Zircaloy-clad fuel may cause unanticipated materials difficulties due to extended exposure to corrosion. This subject should be investigated during the R&D phase.

W3.3-a. Evaluation Against High Level Criteria

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

SBWRs exhibit the following advantages in the area of natural resource utilization:

- Because some of these reactors (e.g., W8 and W23) operate at a lower power density and thus lower fuel temperature than current LWRs, it may be possible to achieve somewhat higher burnup levels resulting in more electric energy and less radioactive waste generated per unit mass of natural uranium. (SU1-1)

- If plutonium-based MOX fuels are utilized, it is possible to significantly increase the amount of electric energy generated per unit mass of natural uranium. (SU1-1). Only the ESBWR lists MOX as a potential fuel. [Note: the use of MOX fuel is not unique to SBWRs.]

On the other hand, because these are thermal reactors, plutonium breeding is not possible and thus the utilization of natural uranium resources is limited compared with fast reactors.

It is concluded that SBWR systems are substantially equivalent to the reference LWRs in the area of fuel utilization.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

Modular SBWRs have the following disadvantage in the area of waste minimization:

- For given installed capacity, multi-module plants are expected to have more activated materials (such as in-pile structures and instrumentation) than large monolithic plants. (SU2-1)

Because spent fuel is the radioactive waste of greatest concern, it is concluded that SBWR systems are substantially equivalent to the reference LWRs in the area of waste minimization.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

SBWRs exhibit the following advantages in the area of proliferation resistance:

- Most of the SBWR concepts are based on the traditional LWR fuel cycle, which has proven to be proliferation resistant over the past four decades. Use of low-enriched uranium and the lack of reprocessing makes diversion of SBWR UO_2 fuel a relatively unattractive path to proliferation. (SU3-1)

- The SBWR concept that proposes the use of thoria-urania fuel would have additional proliferation resistance due to the relatively low production of plutonium and the relatively unattractive plutonium isotopics that are produced. (SU3-1)

- The potentially higher burnup achievable with SBWR oxide fuel would yield end-of-life plutonium isotopics rich in non-fissile isotopes and relatively poor in Pu-239. (SU3-1)

- The long in-pile residence time minimizes the opportunity for fissile material diversion. (SU3-1)

The last two barriers to proliferation can be bypassed by extracting the fuel early in the irradiation cycle, however, this will be relatively transparent. It is concluded that in terms of proliferation resistance, most of the SBWR concepts are substantially equivalent to the reference LWRs. The use of a thoria-urania once-through fuel cycle in W8 makes that concept better than the reference ALSRs.

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

SBWRs exhibit the following advantages in the area of safety and reliability under normal operating conditions:

- Elimination of the recirculation loops and jet pumps increases the overall system reliability.

- Because two of the modular designs operate at a lower power density than current ALWRs, the margin to CHF at steady-state conditions and during transients is larger. (SR1-2)

- Lower temperatures in the fuel may result in better fuel reliability as well as a lower release of fission gases upon fuel pin failure. (SR1-2, SR1-3)

- Re-circulation pump trips are eliminated as accident initiators. (SR1-3)

SBWRs have the following disadvantages in the area of safety and reliability under normal operating conditions:

- For the small modular SBWRs, the monitoring, inspection, and maintenance may be more difficult because of the increased number of components.

The evaluators believe that the SBWRs will probably perform somewhat better than the reference LWRs in terms of safety and reliability under normal operating conditions.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

SBWRs exhibit the following advantages in the area of safety and reliability under accident conditions:

- Elimination of the re-circulation loops and jet pumps increases the system simplicity and reduces the number of pipes that could lead to a LOCA. (SR2-3)

- SBWRs have minimized the need for emergency offsite power and, for at least one SBWR design, the requirement for an emergency AC power supply was eliminated--due to system designs that include fully contained passive cooling systems that rely principally on natural forces (gravity-driven density gradients) and heat transfer to the environment (the ultimate heat sink).

- The ESBWR has a lower core damage frequency that all earlier generation BWRs (SR2-3)

- The SBWRs are designed to have full core coverage by the vessel water inventory under all conditions, using passive safety systems in combination with automatic depressurization (SR2-3). Hence core uncovery, for all scenarios, has been eliminated.

It is concluded that the SBWR concepts will perform better than the reference ALWRs in terms of safety and reliability under accident conditions.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

SBWRs exhibit the following advantage in the area of severe accident mitigation and need for offsite emergency response:

- SBWRs have minimized the need for emergency offsite power and, for at least one SBWR design, the requirement for an emergency AC power supply was eliminated--due to system designs that include fully contained passive cooling systems that rely principally on natural forces (gravity-driven density gradients) and heat transfer to the environment (the ultimate heat sink). (SR3-1)

- For at least one SBWR concept, the requirement for an emergency AC power supply was eliminated (SR3-1).

It is concluded that the SBWR systems will perform better than the reference LWRs in the area of severe accident mitigation and need for offsite emergency response.

Economics–1. Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

SBWRs exhibit the following advantages in the area of operating costs:

- The longer irradiation cycles and the potentially higher reliability of the primary systems should result in higher plant capacity factors. (EC-3)

- The use of natural circulation for normal power operation eliminates the pumping requirements. (EC-3, EC-4)

SBWRs have the following disadvantages in the area of operating costs:

- For the smaller size SBWRs, operation and maintenance of many reactor modules at a single site may result in higher operation and maintenance costs than in current LWRs because of the increased number of components, control rooms, etc. (EC-3)

- To achieve a longer irradiation cycle, some SBWR concepts make use of slightly more enriched uranium than current LWRs. Also, a longer irradiation cycle increases the carrying charges on the fuel. Therefore, the cost of the fuel per unit electric energy generated is expected to be somewhat higher. (EC-3)

- Those SBWR concepts with lower thermal efficiency will also have an even higher fuel cost per unit electric energy generated. (EC-3)

At this point the evaluators believe that it is possible that some and maybe many of the SBWR concepts will perform better than the reference ALWRs in terms of operating costs—however the economic factors have a very high uncertainty. Better cost analysis must be performed for some of the concepts once their designs are completed and their capacity factors evaluated. One major area of cost variability is the influence of plant size on the operating costs.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

SBWRs exhibit the following advantages in the area of capital costs and financial risk:

- Minimal research and development is required to develop the SBWRs because they maximize the utilization of available LWR technology, newly engineered. (EC-4)

- The nuclear island is simplified by eliminating the external-loop piping and several safety-grade systems. (EC-1)

- The design and fabrication approach for modular SBWRs is based on:

  - Factory fabrication—the reactor modules can be fully fabricated in a factory and be readily transported to the site, which reduces expensive on-site assembling/welding, and ultimately, the construction time. (EC-1)

  - Standardization—because a relatively large number of reactor modules will be needed, it will be possible to take full advantage of cost reductions due to learning and standardization. (EC-1)

  - Flexibility—additional generating capacity can be gradually installed at the plant by adding small modules; this will allow the production to match the electricity demand of the utility customers, prevent market saturation, and ultimately maintain a stable price of electricity. (EC-2)

  - Early cash flow—for large plants with many reactor modules, it will be possible to put the first few reactor modules into operation relatively quickly and generate an early cash flow. (EC-2)

  - Lower power density—because the power density is smaller than current ALWRs, the damage to the vessel from fast neutrons should be modest. Therefore, it is expected that the reactor lifetime can be extended beyond that of current ALWRs. (EC-5)

- No pumps will be required since the SBWRs rely on natural circulation for normal operating conditions. (EC-1)

- The ESBWR has a significantly smaller footprint than the ABWR because of the containment redesign. (EC-1)

SBWRs have the following disadvantages in the area of capital costs and financial risk:

- For the modular SBWRs, the smaller power per reactor module and smaller power density within the core result in a larger plant size and amount of materials per unit power generated. (EC-1)

- For a given electric power output, a plant with many reactor modules likely has a larger footprint than a plant with a single large monolithic reactor. (EC-1)

The evaluators believe it is not possible to perform a quantitative economic assessment for the SBWRs and so determine whether they will perform better than the reference ALWRs in terms of capital costs and financial risks. One major area of capital cost uncertainty is the influence of plant size. This concept set includes relatively small modular designs and fairly large monolithic plant designs. It is not clear at this point whether very large or more modest sized plants will be cheapest. In addition, the type of market will be extremely important to that assessment.

W3.3-b. Summary of the Strengths and Weaknesses

Strengths of the SBWR concepts include:

- Relatively modest research and development requirements
- Simplification of the nuclear island
- Higher flexibility in meeting the needs of the electric grid
- Potential for higher plant capacity factors
- Elimination of large LOCAs†
- Passive removal of the decay heat under accident conditions
- No pumps required.

Weaknesses of the SBWR concepts include:

- For the smaller modular SBWRs,
  - More difficult inspection,
  - Smaller power densities, and
  - Larger plant footprint for a given installed capacity.
- Slightly higher fuel costs
- Thermal efficiency at or below current LWR levels

W3.4. TECHNICAL UNCERTAINTIES

W3.4-a. Research & Development Needs

The SBWR, especially the ESBWR is essentially a well-developed concept that can be ready for deployment, with additional engineering and design certification, whenever the market dictates. Therefore, we have not identified any significant research and development for the SBWR as a concept set. However, further evaluation of the economic viability of the modular designs would be appropriate.

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† Due to the elimination of external jet pump lines.

Also, for those SBWR and other concepts with long fuel cycles, some further development of the fuel cladding materials would be appropriate. Also, some development and testing will be need for those concepts that have innovative features such as top driven control rods. And, for those SBWR and other concepts with small modular designs, probabilistic risk analysis to show that, for a given installed capacity, the core damage frequency and dose distribution of a multi-module plant is significantly smaller than that of a single-reactor plant is needed.

W3.4-b. Institutional Issues—Licensability & Public Acceptance

No new and/or specific public acceptance issues were identified for the SBWR concept. The public should be receptive to the elimination of accident initiators by design, to the superior passive safety performance of these systems, and to the minimization of the need for emergency response.

The SBWRs are compatible with the Generation IV proliferation resistance goals.

Licensability of these reactors should be made easier by maximizing the use of existing LWR technology, i.e. fuel, materials and equipment. For those components or systems that are new, it will be necessary to conduct supporting experiments to demonstrate their performance and reliability.

W3.4-c. Timeline for Deployment

Given the relatively small R&D requirements for these reactors, it is expected that the SBWR concepts could be considered for early deployment (before 2015) if conventional fuel is used. If thorium fuel is used, then deployment will likely be after 2015.

W3.5. INITIAL ASSESSMENT: OBSERVATIONS AND CONCLUSIONS

The SBWR reactor concepts make excellent candidates for further assessment. At this point the key issues that will emerge for determining the relative ranking of these systems appear to be their economic values relative to other designs.

W3.6. REFERENCES

Heki, Hideaki, M. Aritomi, M. Nakamaru, “LSBWR1: Long Operating Cycle Simplified BWR,” April 30, 2001.

Ishii, Mamaru, S. T. Revankar, T. J. Downar, and K. Vierow, “Generation IV Simplified Boiling Water Reactor Employing Passive Cooling and Mixed Thorium-Uranium Oxide Fuel Cycle,” Reactor Concept W8, April 16, 2001.

Ishii, Mamaru, S. T. Revankar, and T. J. Downar, “Modular 50 MWe Simplified Boiling Water Reactor Design,” April 16, 2001.

Kataoka, K.,”A Distinctive Coupling Concept of Small Natural Circulation BWR and RO Desalination System (TTBWR+RO),” April 30, 2001.

Khatib-Rahbar, Mohsen, “A Small Advanced Reactor Technology (SMART),” Reactor Concept W7, April 16, 2001.

Rao, A. S., “ESBWR—Bridging the Gap from Generation III to Generation IV,” April 17, 2001.

W3-7. SCREENING SCORE SHEET—SIMPLIFIED BWRS

Summary Evaluation: X Retain Reject

Goal		- -	-	+	++	Comments
SU1	Fuel Utilization			E E		
SU2	Nuclear Waste			E		Longer core life produces fewer SNF packages (marginal benefit)
SU3	Proliferation Resistance					-The UO_2 cores are similar to the reference ALWR
S&R1	Worker Safety and Reliability					Excellent potential for further (but modest) improvement over ALWRs
S&R2	CDF					Excellent potential for further (but modest) improvement over ALWRs
S&R3	Mitigation					Excellent potential for further (but modest) improvement over ALWRs
E1	Life-Cycle Cost					Potential improvements in operating cost, but wide range of plant sizes.
E2	Financial Risk					-ESBWR: smaller footprint than ABWR -Modular SBWRs: among the most proven of Gen IV concepts, but the effect of size on the capital costs is very uncertain

Appendix W4

Pressure Tube Reactors Concept Set Report

December 2002

CONTENTS

FIGURES

TABLES

ABSTRACT

Three advanced pressure tube reactor concepts have been proposed as Generation IV designs. All are based on the commercially successful Canadian Deuterium-Uranium (CANDU) design. The Next Generation CANDU (NG CANDU) concept is a more economic version of the current CANDU design with light water coolant and slightly enriched uranium fuel in conventional CANDU-type bundles. The Passive Pressure Tube Reactor design eliminates heavy water from the calandria and includes a passive core cooling system. It also requires advanced graphite-based fuels. The High-Conversion Pressure Tube Reactor is similar to the Passive Pressure Tube dry calandria design but requires advanced graphite-based fuels with 13.5% ^{235}U driver fuel and mixed Th-U fertile fuel bundles. The advanced pressure tube reactor concepts address the Generation IV goals in that they have significant advantages in the fuel cycle, which enhance sustainability. The passive calandria heat sink provides strong mitigation measure for severe accidents. The NG CANDU option has been optimized to enhance the economics relative to the current ALWR. Capital cost is substantially reduced and the low production cost of the existing CANDU plants is retained. The development costs of the three concepts vary from low for the next generation CANDU design to moderate for the other two concepts, largely because of the extensive fuel development they require.

INTRODUCTION

Pressure tube reactors are a well-established class of water-cooled nuclear reactors in operation around the world. The design of this class is characterized by a heat transport (reactor coolant) system in which the fuel and coolant are subdivided and contained within a set of parallel pressure tubes while the pressure tubes are surrounded by a separate moderator. There have been a number of different pressure tube reactor designs that have been constructed and operated. The most commercially successful of these designs is the Canadian Deuterium-Uranium (CANDU)-type reactor in which both the coolant and the moderator are heavy water (Table 1). In addition to the CANDU designs, the Indian pressure tube reactor program has been successful.

Table 1. Pressure tube nuclear power plants.

Country	Number of Plants	Locations
Canada	18	Multiple sites
Argentina	1	Embalse
Korea	4	Wolsong
China	2	Qinshan (under construction)
Romania	2	Cernavoda (2nd unit under construction)
India	16	Multiple sites
Pakistan	1	Kanupp
Japan	1	Fugen

Several advanced pressure tube reactor design concepts have been proposed as Generation IV reactors. A common feature of these designs is the adoption of light water as the coolant. This design approach is not novel. Two prototype pressure tube reactors have been built with light water coolant. The Winfrith Steam Generating Heavy Water Reactor (SGHWR), which went into operation in England in 1968 and was shutdown in 1990, was a 100 MWe direct-cycle design with light water boiled in vertical pressure tubes surrounded by low-pressure heavy water moderator. The SGHWR was constructed to prove the design technology with both boiling and superheat fuel channels and included separate experimental loops. This design was not commercialized because its prospective utility customers at the time (in England and Scotland) judged the remaining development costs as too high and opted for alternative commercially available reactor designs. A similar vertical pressure tube reactor design was built in Japan and commissioned in 1979. The Fugen reactor is 165 MWe direct cycle, vertical pressure tube reactor with a heavy water moderator. There are numerous technical differences in the designs of these two reactors, but together they have proven the feasibility of the light water coolant/heavy water moderator combination for pressure tube designs.

Three new pressure tube reactor (PTR) designs have been proposed as Generation IV concepts as listed in Table 2. All of these concepts differ from the SGHWR and Fugen designs by having the pressure tubes oriented horizontally in order to take advantage of on-line refueling and they employ an indirect steam cycle. They can all be considered as advances on the CANDU-type reactor design. The key differences in the proposed concepts are in the moderator/calandria design and the fuel design.

Table 2. Generation IV pressure tube reactor concepts.

Concept	Key Features	Sponsor
Next Generation CANDU (NG CANDU)	Light-water coolant Heavy-water moderator in calandria Slightly-enriched uranium fuel	AECL
Passive Light-Water Pressure-Tube Reactor (Passive PTR)	Light-water coolant Option 1: No separate moderator - Gas-filled calandria and graphite reflector, CANDU-type fuel Option 2: Light-water moderator & graphite matrix fuel	MIT
High Conversion Pressure Tube Light Water Reactor (High Conversion PTR)	Light-water coolant Light-water moderator Gas-filled calandria Thoria-urania fuel	Kyung Hee University

The primary drivers of the three concepts are different. The main driver for the advances in the next generation CANDU design is improved economics, achieved principally through a capital cost and schedule reduction. Key features that enable the improved economics are a reduction in the heavy water inventory, an increase in thermal efficiency, a smaller core, and a design based on modular construction. The Passive Pressure Tube Reactor (Passive PTR) design is focused on passive safety design while the High Conversion PTR design is focused on fuel cycle optimization.

Concept Description

Table 3 summarizes the main design parameters for advanced pressure tube reactors and the current generation CANDU 6. Complete design details are not available for all of the proposed variations on the pressure tube design. The thermal-hydraulic characteristics of the advanced pressure tube reactor concepts are an extension from the current CANDU designs to deliver higher thermal efficiency. Brief description of the concepts is given in the following sections. More details are available in the Generation IV concept submissions and the references. The developers of the concepts primarily wrote the concept summaries reported below. They have been edited for style and brevity. Some of their statements may not reflect the judgment of the Technical Working Group, which is reported instead in Section 3 of this appendix.

Next Generation CANDU (W6)

The next generation CANDU design (Figure 1) is based on the standard CANDU design with horizontal pressure tubes fuelled on-line with short fuel bundles and surrounded by a low-temperature heavy water (D2O) moderator.[a] The CANDU design features include a high neutron efficiency, ease of construction and localization. An inherent safety feature of the design is a passive moderator/shield tank heat sink surrounding the pressure tube core. The major innovations in the next generation CANDU are:

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a. Duffey, R. B. et al. 2000; Bushby, S. J. et al. 2000; Wren, D. J. et al. 2001; Hopwood, J. .M. et al. 2001; Hau, K. F. et al. 2001; and Chan, P. S. W. et al. 2001.

Table 3. Pressure Tube Reactor concept technical data.

	Next Generation CANDU (W6)	Passive PTR (W28)*	High Conversion PTR (W5)	CANDU 6**
Net Power Output	400-1200 MWe (650 MWe***)	>1000 MWe	na	700 MWe
Reactor Core number of fuel channels	256 (dependent on unit output)	740	380	380
average channel power	~ 6.5 - 8 MWth	5.4 MWth	na	5.8 MWth
core diameter	~ 5 m	8.7 m	na	7.6 m
fuel channel	Zr-2.5%Nb	Zr-2.5%Nb	na	Zr-2.5%Nb
lattice pitch	220 mm	286 mm	na	286 mm
Fuel	43-element CANFLEX uniform ~1.6% ^{235}U Thorium option	Dry – graphite matrix Wet – 24 element bundle 2% ^{235}U	Graphite Matrix 13.5% ^{235}U driver fuel ThO_2 + 5%^{235}U fertile fuel	37-element uniform natural uranium
Operating Parameters outlet temperature	~ 330°C	338°C	Na	310°C
outlet pressure	~ 13 Mpa	14 MPa	na	10 MPa
Heat Transport System Steam Generators	2 – vertical U-tube with integral preheater	na	na	4 – vertical U-tube with integral preheater
Heat transport pumps	4 – vertical, centrifugal	na	na	4 – vertical centrifugal
Containment Type	Pre-stressed concrete	Passive cooling	na	Pre-stressed concrete epoxy
liner	Stainless steel	Stainless Steel		
Turbine Generator	Single flow, high-pressure cylinder and double flow, low-pressure cylinder	na	na	Single flow, high-pressure cylinder and 2 double flow, low-pressure cylinder
Gross Electrical Efficiency	>36%	na	na	35%
Capacity Factor	90%	na	na	85%

na = No design details available
* Based on Tang et al. 1994.
** Current Generation product for comparison.
*** Submitted design.

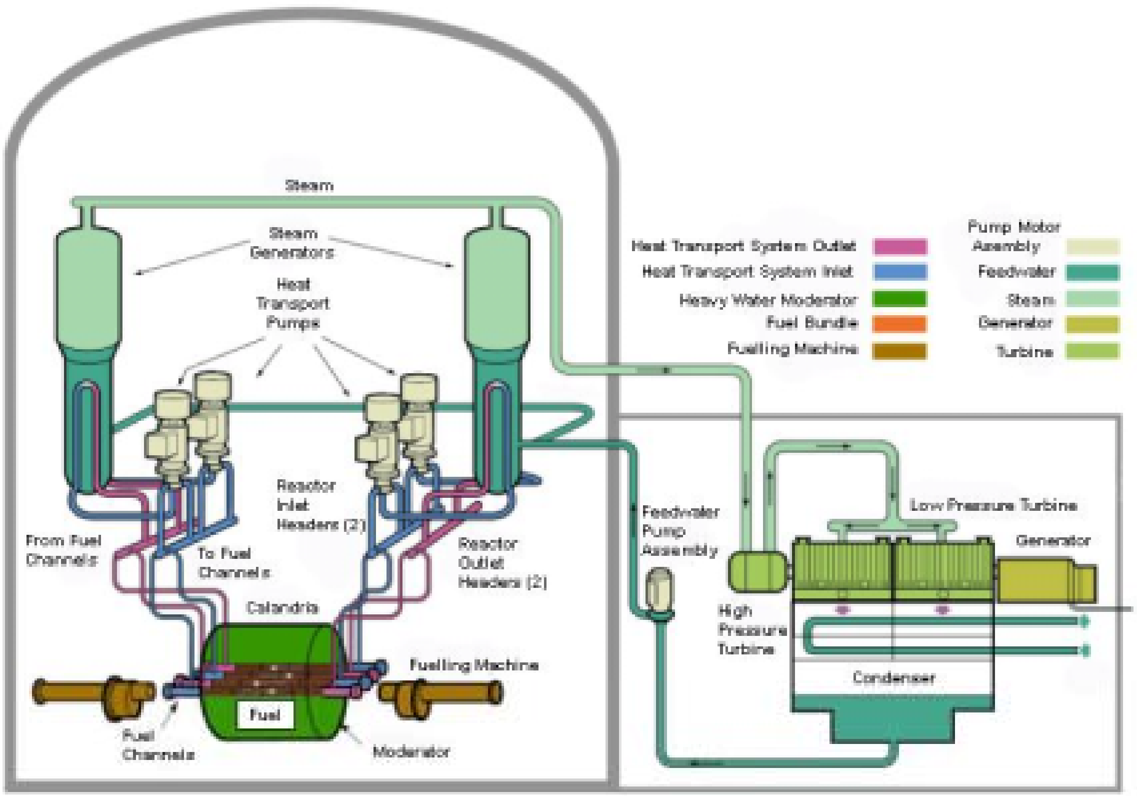

Figure 1. Overall Next Generation CANDU plant flow diagram.

- More compact core design
- Replacement of heavy water in the reactor coolant system with light water
- Slightly enriched uranium oxide fuel in CANFLEX fuel bundles
- Higher thermal efficiency
- Enhanced passive safety systems
- Improved performance through advanced operational and maintenance information systems.

The performance of the next generation CANDU designs will be improved through an optimization of the reactor core configuration. It is possible to design a highly efficient core that maximizes the ratio of power to heavy water. This results in a more compact reactor core, a smaller calandria vessel and optimized reactor internal components. The internal dimensions of the CANDU pressure tube are retained in order to ensure advances in fuel bundle design are interchangeable and applicable to the full range of CANDU systems. A much more compact core and the elimination of the heavy water requirement in the reactor coolant system sharply reduces the inventory of heavy water in the moderator, which results in a major cost reduction for the next generation designs.

The next generation CANDU reactor is designed to use slightly enriched uranium fuel and light water coolant. Similar to other CANDU designs, next generation CANDU will have an efficient heavy water moderator, low neutron absorbing zirconium alloys for the core structures, fuel cladding and horizontal fuel channels that contain the fuel.

The use of small diameter fuel channels to contain high pressure, high temperature reactor coolant allows the use of a separate low-pressure moderator system in which the reactivity control devices operate. The core uses on-power replacement of fuel to maintain sufficient positive reactivity. This feature contributes to high availability factors and outage flexibility since refueling outages are not required in CANDU reactors.

The basic arrangement of the reactor (Figure 2) consists of a cylindrical calandria and end shield assembly supported by a cylindrical shield tank. The calandria contains heavy water moderator; the shield tank contains light water, which serves as both a thermal and a biological shield.

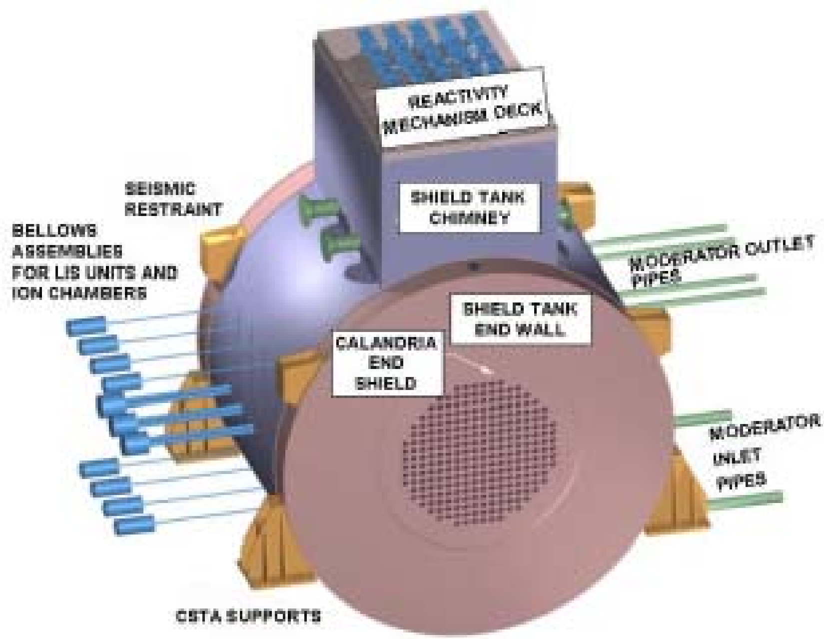

Figure 2. Next Generation CANDU reactor assembly.

The lattice sites are arranged in a square array, parallel to the horizontal axis of the calandria. Each of the lattice sites is occupied by a fuel channel assembly, which passes through the calandria. There are 256 fuel channels in the reference core, each containing 12 fuel bundles. The fuel channel consists of a zirconium-niobium pressure tube, centered in a calandria tube and expanded into a stainless steel end fitting at each end. The annulus between the pressure tube and the calandria tube is gas-filled to provide thermal insulation between the hot coolant and the relatively cool moderator. Spacers positioned along the length of the pressure tube prevent contact between the two tubes.

The calandria is comprised of a cylindrical shell and with flat end shields at each end. Each end shield is made up of two tube sheets joined by lattice tubes and a peripheral shell. The space between the end shield tube sheets is filled with steel balls and water for shielding. This shielding allows personnel access to the reactor face during reactor shutdowns. The shield tank is a cylindrical vessel that is concentric around the calandria.

The Nuclear Steam Supply System (NSSS) (Figure 3) of the next generation CANDU is similar in concept to the standard CANDU, however the light water reactor coolant allows the auxiliary systems to be simplified and some to be eliminated. The increase in the reactor coolant system pressure and also the steam system pressure allows a more compact steam turbine to be utilized and provides an overall increase in thermal efficiency.

The two safety shutdown systems designs are similar to those used in the standard CANDU design, and have been retained. The emergency cooling system design is significantly simplified and improved through the use of the light water coolant. The containment is based on a pre-stressed concrete design with a steel liner. Options for passive cooling of the containment are being considered. The emissions, from the plant during normal operation and under postulated accident conditions would be significantly reduced.

The more compact core offers improvements in manufacture, installation and also allows the reactor building size to be reduced. This and the simplification of the NSSS allow the reactor building to be reduced by at least 10% below the standard CANDU plants. Similarly, the improvements in the balance of plant and the future generation design allow a significant reduction in the overall plant footprint. Improved design methods and construction techniques, developed on the standard CANDU products, are being implemented into the design at an early stage to ensure the design is optimized to meet the cost and schedule targets set for the next generation CANDU products.

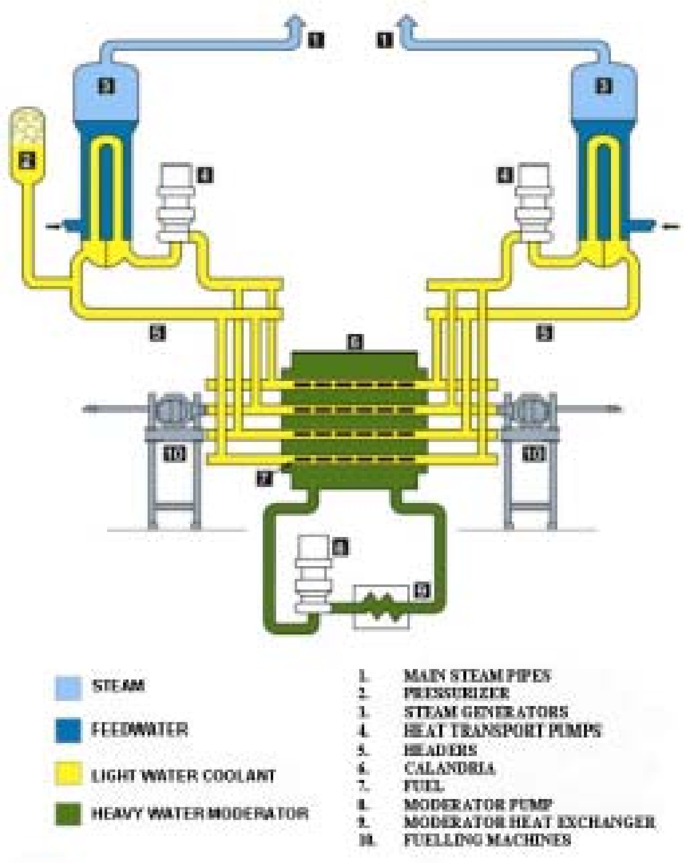

Figure 3. Next Generation CANDU nuclear steam supply system**.**

A next generation CANDU concept design has been developed for a 650 MWe plant. However, unit output can be varied from 400 to >1200 MWe (Figure 4) to meet market needs for flexibility and capital outlays. Modular techniques are used in design and construction to reduce cost, enable rapid construction, and ensure full safety and quality assurance while still meeting international and national licensing requirements.

The extensive application of probabilistic safety assessments during the design phase, supported by the CANDU industry experience base, is leading to designs that reduce accident risks and meet ALARA goals. The reliability of the NG CANDU is projected to be better than the current LWRs due to advances on current CANDU computerized control and instrumentation.

The target for this design is a capital cost reduction of 30–40% compared to current CANDU or the best LWR designs. The design is being optimized for a rapid project implementation schedule of 48 months for the Nth units.

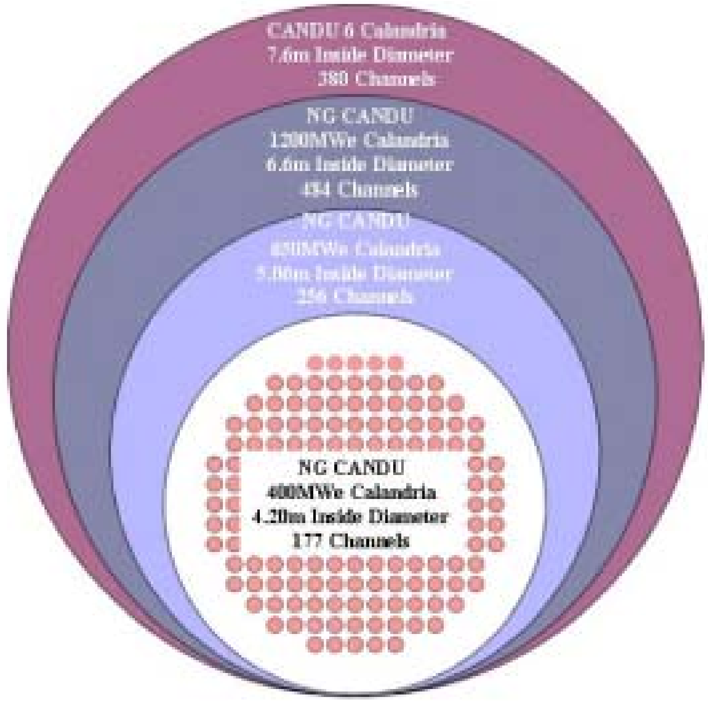

Figure 4. Comparison of Next Generation CANDU Core Sizes

Passive Pressure Tube Reactor (W28)

The Passive Pressure Tube Reactor is a high-power (> 1000 MWe) design, which includes a number of features to maximize the thermal margins for fuel performance and to prevent fuel failures (e.g., graphite-based high-temperature-resistant fuel). It also includes features to optimize the ability of the pressure tubes to dissipate decay heat to the moderator (e.g., calandria flooding) thereby increasing the level of passive safety protection in the general design. Details on the design concept and design features are given in the References (Tang, et al. 1994; Hejzlar, P. et al. 1993a; Hejzlar, P. et al. 1993b; Hejzlar, P. et al. 1993c; Hejzlar, P. et al. 1995; Hejzlar, P. et al. 1996a; Hejzlar, P. et al. 1996b; Hejzlar, P. et al. 1997; Hejzlar, P. et al. 1998a; Hejzlar, P. et al. 1998b; Kim, M.H. et al. 1997; and Tang, J.R. et al. 1994)

The Massachusetts Institute of Technology (MIT) has proposed two variants of the Passive PTR concept (Figure 5). Both designs are based on the current CANDU reactor design. The key differences are the design of the calandria and fuel, and the elimination of the Emergency Core Cooling System (ECCS).

The Dry Calandria version has no moderator on the outside of the fuel channels. The light water coolant provides the required moderation and there is a solid graphite reflector inner liner to the calandria. Under normal operation the calandria space is filled with a low-pressure gas in balance with a water column in the containment building. In the event of a loss of coolant accident, the calandria is flooded (actuated by a passive valve) and long-term decay heat removal is ensured by heat loss from the pressure tubes to the large volume of water available to flood the calandria. The fuel for the Dry Calandria version is SiC-coated graphite matrix with coolant channels and TRISO particles in fuel compacts. The SiC coating is required to protect the graphite from oxidation in high temperature steam. Analyses show that this design is capable of dissipating heat from voided fuel elements without exceeding design limits.

The Wet Calandria version also has a gas-filled calandria vessel like that in the Dry Calandria version, but without the flooding capability. The fuel channel for the Wet Calandria version includes a thin-walled Zircaloy tube, which creates an annular space around the calandria tube that is filled with low-pressure low-temperature light water moderator. This annular moderator acts as heat sink during both normal operation and during loss of coolant events. Heat from the moderator is dissipated passively to the containment atmosphere by natural circulation to reservoirs located on the calandria wall (Figure 6). The fuel for the Wet Calandria version is a multi-pin fuel bundle, similar to the CANDU bundle design, but with a SiC-coated graphite plug replacing the center pin and with the traditional Zircaloy fuel cladding replaced by SiC cladding or another corrosion resistant ceramic. The Wet Calandria version has a relatively flat thermal flux profile, negative coolant and moderator void coefficients and tight neutronic coupling.

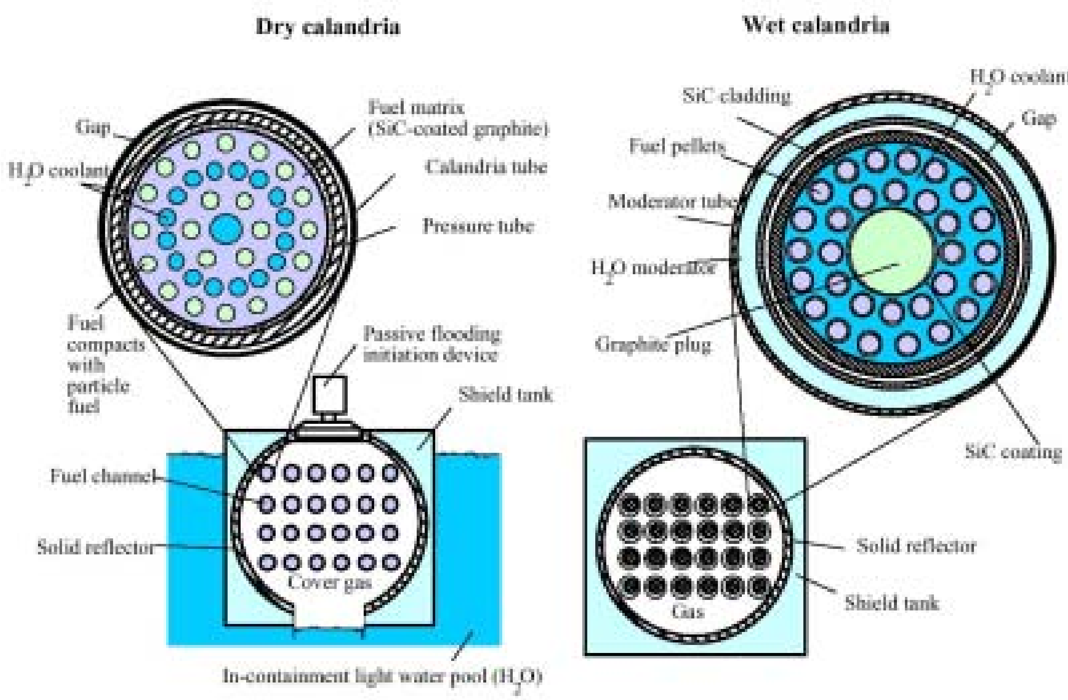

Figure 5. Wet and dry calandria Passive Pressure Tube Reactor's fuel channels.

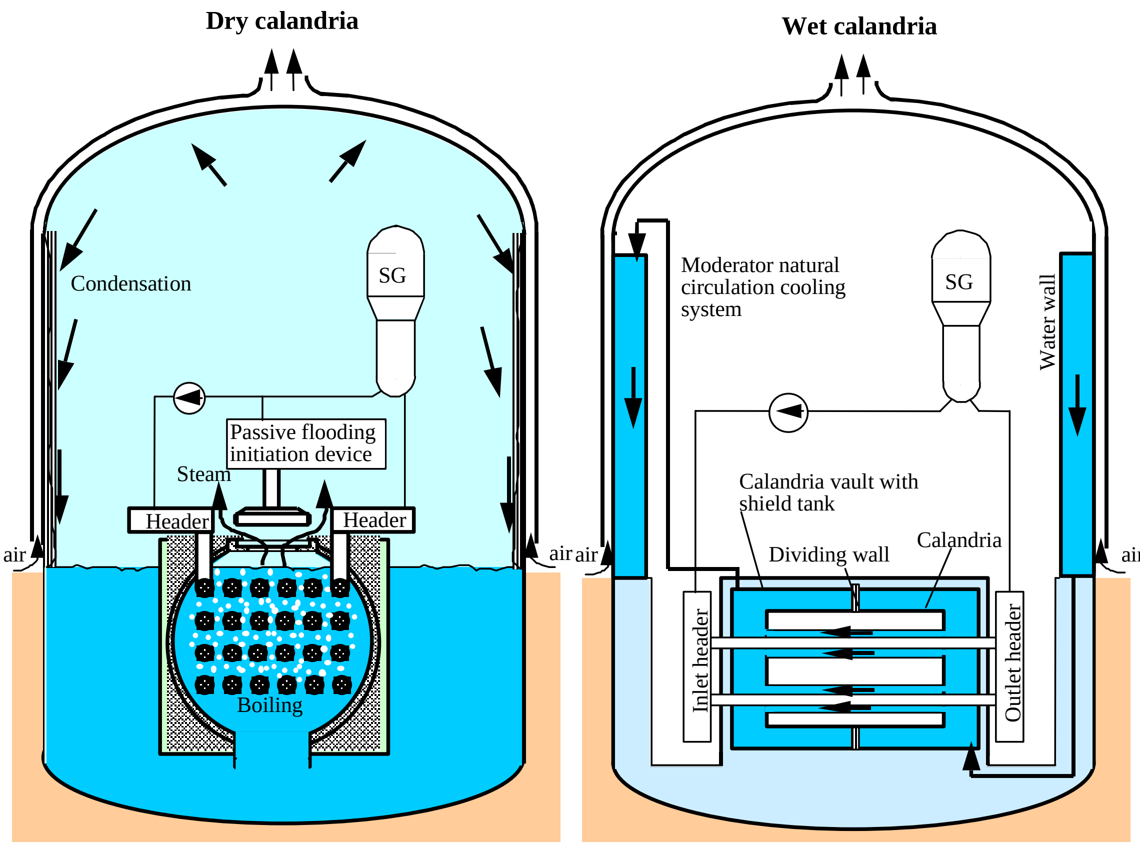

Figure 6. Ultimate heat removal path for the dry and wet calandria versions.

High-Conversion Pressure Tube Reactor (W5)

The High Conversion PTR is similar in design to the Dry Calandria version of the Passive PTR, but there are only limited details on the proposed overall plant design (Kim, M. et al. 1999; Kim, M. et al. 2000). Like the Passive PTR, the High Conversion PTR has a gas-filled calandria surrounding the horizontal pressure tubes. For this design, flooding of the calandria under accident conditions is achieved passively by gravity feed from a light water reservoir located above the calandria.

The fuel for the high Conversion PTR is a once-through thorium-uranium seed and blanket type fuel. The overall dimensions of the fuel bundles are the same as for normal CANDU fuel, however, to maximize the conversion ratio, the fuel pin diameters are smaller and the pins are bundled with a tighter pitch. The seed fuel is placed in every fourth pressure tube and consists of 13.5% ^{235}U in a uranium-15%Zr metal matrix. The blanket fuel is BISO coated thoria (ThO_2) and 5% ^{235}U uranium oxycarbide (UCO) particles embedded in a graphite matrix. Both the seed metal fuel slugs and the blanket pressed and sintered graphite matrix pellets are clad with Zircaloy. Channels are fuelled at a ratio of one seed channel to three blanket channels. The blanket fuel kernels and the seed and blanket enrichments are designed for a blanket fuel residence in the core of 10 years and for leveling of the power density between the seed and blanket channels.

Potential to Meet Generation IV Goals

In the following subsections, the Pressure Tube Reactor concept set is assessed against the Generation IV goals. The advantages and/or disadvantages of this concept set are evaluated relative to a typical Generation III ALWR reactor. In those areas for which no appreciable differences can be identified between the Pressure Tube Reactor concept set and the references, the analyzed concept is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and metrics by means of a label in parentheses.

Evaluation Against High Level Criteria

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

Pressure tube reactors have the following advantages relative to the reference reactors with respect to Sustainability-1:

- The use of heavy water in the moderator of the next generation CANDU leads to a fuel utilization greater than 7.8 MWd/kg natural uranium (NU) extracted. This is better than the energy efficiency of current natural uranium-fuelled CANDUs (7.5 MWd/kg NU) and typical light water reactors with 3.5% enriched fuel with 40 MWd/kg U burn-up (6.2 MWd/kg NU). (SU1-1)

- The PTR designs with graphite matrix fuel offer higher burn-up potential and provide better uranium utilization than current PWRs. (SU1-1)

- All PTR designs are capable of operation with alternative fuel cycles including thorium fuel cycles. The High Conversion PTR with the mixed thoria fuel design has the potential to extend resource sustainability. (SU1-1)

- The PTRs are able to burn spent LWR fuel subjected to dry recycling (DUPIC) without the need to add additional fissile material, thereby avoiding the mining of additional ore. (SU1-1)

Based on the above factors, pressure tube reactors are assessed to be better than the reference ALWRs.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

Pressure tube reactors have the following advantages relative to the reference reactors with respect to Sustainability-2:

- The neutron economy of the next generation CANDU design makes it suitable to burn recycled LWR fuel. This offers the potential to reduce overall waste volumes from a combination of LWR and PTR plants using the DUPIC fuel cycle. (SU2-1, SU2-3)

- Adoption of an advanced fuel as proposed by the High-Conversion PTR would significantly reduce the waste volume. (SU2-1)

Pressure tube reactors have the following disadvantages relative to the reference reactors with respect to Sustainability-2:

- Use of slightly enriched fuel and higher burnups will significantly reduce the volume of spent fuel from the pressure tube reactors compared to current CANDU plants, but the PTRs will still produce a larger high-level waste volume (about twice as much) than the reference ALWRs per MWd. However, the total actinide and heat loading of the spent fuel will be about the same as the reference ALWR except for the high conversion option. For this option, the residual actinide levels will be lower for the total energy produced. (SU2-1)

- Reduction in the heavy water inventory and the use of a light water coolant will significantly reduce the tritium production and potential emissions from the PTRs, however, the tritium production will be somewhat larger than the reference ALWRs and the release to the environment may be greater than the reference, which also releases tritium. (SU2-2)

Overall, Pressure Tube Reactors are assessed to be better than the reference ALWRs when the DUPIC and high conversion fuel cycles are used. The once through cycles are assessed to be moderately worse than current ALWRs due to a larger waste volume.

It should be noted that the long-term stewardship burden of the pressure tube reactors would be essentially similar to that of the reference plants. The requirements for management and disposal of zirconium-clad uranium oxide fuels are well established and technically proven options for disposal in geological repositories are available. The long-term management and disposal of graphite matrix fuels is less well established. It is likely to result in a similar stewardship burden, but research is required to confirm this. The use of recycled LWR fuel in the next generation CANDU offers the potential to reduce the net actinide and plutonium burden in a geological repository for spent fuel.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

Pressure tube reactors have the following advantages relative to the reference reactors with respect to Sustainability-3:

- The PTR designs are based on the use of only slightly enriched uranium fuel (1.5-2%), which limits the need for the production of higher-level enrichments that could be more attractive to divert. (SU3-1)

- The next generation CANDU plant will be able to operate using DUPIC (**D**irect **U**se of **P**WR fuel **I**n **C**ANDU) fuel. Since this fuel can be manufactured using a dry processing technique that does not separate the fissile material from most of the fission products, the recycling process is not subject to diversion of fissile material. (SU3-2)

- The PTR design offers the option of a thorium fuel cycle with dry spent fuel processing, which is inherently proliferation resistant. (SU3-3)

Pressure tube reactors have the following disadvantages relative to the reference reactors with respect to Sustainability-3:

- All PTR designs include on-line refueling. This has the disadvantage of providing an opportunity for fast fuel shuffling to produce plutonium with a relatively large fraction of PU-239 and Pu-241. Safeguards systems have been designed and adopted at operating plants that have proven adequate to prevent such diversion to date. Once irradiated, PTR fuel is less attractive for diversion than the fuel of current plants owing to the lower fissile content. (SU3-2)

- The DUPIC recycle involves transportation to the recycling facility and this adds a minor increase to the diversion potential. (SU3-2)

- The High-Conversion PTR requires up to 13.5% enrichment and is consequently less proliferation resistant than the reference once through low enriched fuel cycle. (SU3-1)

Pressure tube reactors are assessed to be moderately worse than current ALWR.

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

There are no unique worker safety issues with the advanced PTR designs. Routine exposures from PTR designs should be similar to those of the reference ALWR plants (some factors such as tritium are larger, other factors are smaller). Knowledge and experience from current plant design and operation can be used in the advanced designs to control radiation exposures and limit routine releases to the public. The use of light water in the heat transport system will sharply reduce the levels of tritium production and release from PTRs compared to the current CANDU plants. (SR1-2)

The reliability of PTR reactors should be similar to the reference plants. In general, the equipment and designs of the systems outside of the reactor core will be similar to those of current CANDU plants. The advanced designs will include layout provisions to facilitate maintenance and equipment redundancy to reduce down time. Lessons learned from current plants on maintenance needs should lead to enhanced capacity factors.

Pressure tube reactors have the following advantages relative to the reference reactors with respect to Safety and Reliability 1:

- On-line refueling provides a means of quickly removing failed fuel elements to limit fission product inventories in the heat transport system. (SR1-2)

- Both wet and dry calandria designs provide space for the deployment of more detectors and control devices in the core. The dry calandria concepts have long neutron free paths and a large neutron migration area. Tight core coupling enhances the effectiveness of core monitoring and control. (SR1-3)

- The combination of containment building access for maintenance during normal operation and on-line refueling could give the PTR designs an advantage in terms of reliability. (SR1-1)

Pressure tube reactors have the following disadvantages relative to the reference reactors with respect to Safety and Reliability 1:

- In general, the LWRs are supported by many institutional initiatives, which have led to increased performance in the USA. Support for PTRs is less mature in many jurisdictions and performance improvement initiatives were only recently started. (SR1-1)
- Because heavy water is used to moderate the core, the worker exposures may increase due to the additional tritium that will be produced compared to the reference ALWRs. However, it should be noted that the CANDU experience shows that worker exposure at heavy water moderated and cooled plants can be reduced to acceptable levels. (SR1-1)

Therefore, pressure tube reactors are assessed to be equivalent to the reference ALWRs.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

The system model uncertainties for the advanced PTRs should be similar to those of current plants. A suite of analysis tools is available for CANDU reactors that would be applicable to all of the proposed PTR designs. An exception is the area of heat transfer from the advanced graphite fuel matrix designs that have been proposed.

Pressure tube reactors have the following advantages relative to the reference reactors with respect to Safety and Reliability 2:

- The concepts include unique graphite-matrix fuel designs and fuel cycles. The fuel design, incorporating TRISO fuel particles, offers an increased robustness in terms of fuel integrity and the prevention of fission product release for a wide range of accident scenarios. (SR2-3)

- The NG CANDU design has been optimized using a detailed reliability-centered maintenance assessment. Infrequent single failures that can lead to plant outages have been eliminated. (SR2-3)

- There is no possibility of control rod ejection accidents because the control rods do not penetrate the high-pressure reactor coolant boundary. (SR2-3)

- The PTR concepts include dual shutdown systems, which reduce the frequency of core damage. (SR2-3)

Pressure tube reactors have the following disadvantages relative to the reference reactors with respect to Safety and Reliability 2:

- The designs of the Passive PTR and the High-Conversion PTR have not been subjected to a full safety analysis and there are uncertainties in the behaviour of the fuel channels and heat removal rates under the full range of potential accidents that need to be addressed. (SR2-2)

- The next generation CANDU design has a small negative power coefficient and may have a small positive void coefficient. This design feature is accommodated by the presence of two independent and diverse fast-acting shutdown systems (more than the ALWR designs) to prevent a reactivity transient in the event of a large loss-of-coolant accident. The next generation CANDU design could alternatively achieve a negative void coefficient, if required, through a small adjustment in the fuel design. The designs of the other concepts in this group do not have a positive void coefficient. (SR2-3)

Pressure tube reactors are assessed to be similar to the reference ALWRs.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

Pressure tube reactors have the following advantages relative to the reference reactors with respect to Safety and Reliability 3:

- Compared to the LWR design, all of the PTR designs include the availability of an extra heat sink in the calandria that is separate from the heat transport system. This heat sink will either prevent or significantly retard the progress of a severe accident. In the next generation CANDU design, this heat sink is always present as a heavy water moderator, while for the other concepts this heat sink is made available by actuation of calandria flooding. (SR3-1)

- The advanced PTRs offer the potential for reduced risk if the passive features of the proposed concepts are adopted. The combination of graphite matrix/TRISO fuel plus calandria flooding could potentially eliminate the possibility of a large radioactivity release during any hypothetical accident. (SR3-4)

- The next generation CANDU includes advances to increase the robustness of the current CANDU design including thicker pressure tubes, fuel with improved thermal margins and more reliable safety systems. (SR3-1)

Pressure tube reactors are assessed to be better than the reference ALWRs.

Economics–1. Generation IV nuclear energy systems will have a clear life cycle cost advantage over other energy sources.

Pressure tube reactors have the following advantages relative to the reference reactors with respect to Economics-1:

- Current CANDU reactors have very low operating costs (e.g., the Darlington station has a production cost of about 1 cent U.S./kWh) and the NG CANDU will build on this base. The low fuel costs of the CANDU are the main factor in the low production cost. The switch to slightly enriched uranium for NG CANDU does not change the fuelling cost ($/KWh). (EC-3)

Pressure tube reactors have the following disadvantages relative to the reference reactors with respect to Economics-1:

- The fuel costs for the Passive PTR and the High Conversion PTR may be higher than for the reference once-through ALWR fuel.

The NG CANDU Pressure tube reactor is assessed to be significantly better than the reference ALWRs. The Passive PTR and the High Conversion PTR are assessed to be somewhat worse than the reference ALWRs.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

The financial costs of a plant depend on the unit design and size and the project implementation schedule. The sizes of the proposed concepts range from small to large so that there will be a commensurate range in the financial costs.

Pressure tube reactors have the following advantages relative to the reference reactors with respect to Economics-2:

- The projected capital cost of the NG CANDU is expected to be significantly lower than for current ALWR and CANDU plants (about $1000U.S./KW). This cost estimate is based on the expected cost savings associated with the reduction in the heavy water inventory and the size of the calandria, and improvements in plant thermal efficiency. The cost increases associated with the passive features of the other concepts are not known so that there is considerable uncertainty in the overall capital cost reduction for this group of concepts as a whole. (EC-2)

- The financial risk in building a NG CANDU is addressed by several factors. Since the concept is based on the proven CANDU energy systems, regulatory risk is minimized. The concept is economic at a relatively small unit size (600 MWe or smaller depending on market conditions) and the concept can be readily scaled to even smaller (or larger) unit sizes, allowing the capital requirement to meet risk allowance. Finally, the design of the next generation CANDU may be optimized to permit a rapid project implementation schedule. (EC-2)

- The target for the next generation CANDU plant is a 48-month total schedule (contract-effective-date to in-service), which is significantly better than current plants. (EC-2)

- The development costs of the advanced PTRS will cover a range from low to moderate. All of the proposed concepts are based on the current CANDU designs for all of the systems outside of the reactor core, with the exception of special passive heat removal systems. The design of the next generation CANDU, in particular, is a relatively modest extension of the design of the currently operating CANDU plants. The development of this concept carries the lowest risk and will be relatively modest. (EC-4)

- The profitability of the advanced PTRs will be higher than that of the reference plants if the targets for capital cost reduction can be achieved and the targets for unit energy production can be achieved. (EC-5)

Pressure tube reactors have the following disadvantages relative to the reference reactors with respect to Economics-2:

- Like many of the Generation IV concepts, the proposed PTRs are still in an early development phase. While the economic targets of the NG CANDU represent a significant improvement over current plants, attaining these targets needs confirmation. (EC-1)

- The other two PTRs are at a preliminary concept development stage and the design objectives of these concepts were not primarily reduced cost. Therefore, the economics of these concepts might be worse than the current plants. (EC-1)

- The development costs of the Passive PTR and the High-Conversion PTR may be significantly higher where there are divergences from the established CANDU design, particularly in the area of the fuel channel and the fuel design. (EC-4)

Based on the NG CANDU design, pressure tube reactors are assessed to be significantly better than the reference ALWRs. The capital costs associate with the Passive PTR and the High Conversion PTR are uncertain.

Strengths and Weaknesses

The potential of the PTR concept is summarized in Table 4. Overall, the concept is a very good candidate for further development.

Table 4. Concept strength and weakness.

Category	Strength	Weakness
Sustainability	Improved neutron economy and fuel utilization Potential for advanced fuel cycles such as DUPIC Proliferation resistance for once-through Th fuel cycles	Thoria fuel cycles need further development Graphite-matrix fuel designs need development On-line refueling is a potential proliferation concern
Safety	Lower core damage frequency More passive heat removal capability with flooded calandria Greater fuel thermal margin with graphite-based fuel options Slower severe accident progression	Positive void coefficient Tritium inventory in heavy water moderator Safety technology for Passive PTR design needs to fully developed
Cost	Lower capital cost Lower financing costs and improved profitability Low development cost – based on proven design Faster project schedule Low generation costs relative to ALWR	Uncertainty in product delivery schedule Capital cost reduction dependent on unit size and realization of the cost reduction targets

Technical Uncertainties

Research and Development Needs

The pressure tube reactor concepts are a direct evolution from the operating CANDU units. As such, the majority of the systems and components in the plant, plus the overall design concept, are proven and based on existing technologies and available components. However, the various PTR concepts include several areas where technology development is required. The development needs are different for the different concepts.

The next generation CANDU concept is based on current technology, so that the overall investment to bring the design to the state ready for deployment is low and the technical risk is also low. For both the Passive PTR and the High-Conversion PTR, the development costs and risks will be higher, particularly in those design areas that diverge significantly from the base CANDU design. Those will include validation of the fuel design and performance and validation of the passive safety features of the designs.

Improved thermal efficiency requires fuel channel operation at higher temperatures and pressures than used in the current CANDU designs. The use of enriched fuel in the advanced PTR concepts allows for an increase in the pressure tube thickness to accommodate the increased system loads. However, a program will be required to qualify the performance of the pressure tubes for the new service conditions.

For all PTR concepts, the target of capital cost reduction and reduced construction schedule require an optimization of the plant design. Adoption of improvements available in the areas of electronic and communications technology to plant design can be a significant contributor to achieving this target.

Table 5 summarizes the major areas where technical development is required. The development rating is given for a combination of cost and risk.

Table 5. Pressure Tube Reactor development requirements.

Design Area	Requirement	Development Rating	Concept
Fuel Development	Validation of CANFLEX slightly enriched uranium	Low	NG CANDU
	Thorium fuel cycle	Moderate	NG CANDU
	Graphite matrix thorium fuel	High	H-C PTR
Safety Analysis	Validation of analysis tools	Low	NG CANDU
		Moderate	H-C PTR, PPTR
	Validation of core physics design	Moderate	H-C PTR, PPTR
Fuel Channel	Qualification of channel components for higher temperature and pressure	Moderate	All concepts
Reactor Assembly	Qualification of reactivity control design	Low	NG CANDU
		Moderate	H-C PTR, PPTR
Fuel Handling	Qualification testing of fuelling machine	Moderate	All concepts
Passive Systems	Validation of passive system design	Low	NG CANDU
		Moderate	H-C PTR, PPTR
Control and Instrumentation	Optimization of control and instrumentation design	Low	All concepts

Institutional Issues—Licensability and Public Acceptance

There are no difficult licensing issues associated with this concept. The pressure tube reactor has been licensed for operation in many countries around the world. Pressure tube reactor designs have been considered in the past for construction in the United States and preliminary assessments were that they could be licensed (Shapiro, N.L. et al. 1979). The design concepts meet public goals for increased plant safety. There could be public acceptance issues associated with the proliferation resistance of on-line refueling, but these can be addressed with an adequate safeguards program.

This design option addresses public goals for increased sustainability in terms of fuel utilization, particularly if a thorium fuel cycle is adopted.

There may be a public acceptance issue associated with a positive void coefficient in a pressure tube. Variants on the fuel design of this concept can deliver a negative void coefficient if this is necessary.

The graphite-based fuel designs should deliver increased margin against fission product release in the event of accidents, which is attractive for licensing and public acceptance. However, these advance fuel designs require considerable development and there may be public concerns with fuel cycles that require fuel enrichments above 5% ^{235}U.

Time Line for Deployment

Given the relatively small R&D requirements for the NG CANDU, it is expected that this reactor can be deployed before 2015. The other PTR concepts require more R&D and definition and are expected to be deployed later in the GEN IV period (before 2030).

Statement of Overall Concept Potential

The pressure tube reactor concepts address the Generation IV goals in that they have significant advantages in the fuel cycle, which enhance sustainability. The passive calandria heat sink provides strong mitigation measure for severe accidents. The NG CANDU option has been optimized to greatly enhance the economics relative to the current ALWR. Capital cost is substantially reduced and the low production cost of the existing CANDU plants is retained.

The characteristics of the advanced pressure tube reactor designs make them good candidates for further assessment. The issues associated with development and commercialization of an advanced pressure tube concept appear manageable. The uncertainties lie in the degree to which the designs can achieve economic and safety targets and not in whether the design is capable of surpassing the current plant designs.

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Top-Tier Screening Table

Concept: __________PRESSURE TUBE REACTORS____

Summary Evaluation: _X___ Retain ____ Reject

	Goal	--	-	+	++	Comments
SU-1	Fuel Utilization					Uses 25% less uranium than ALWR & also can use DUPIC
SU-2	Nuclear Waste					-The PTRs (including the NG CANDU) with a once through fuel cycle produce about twice as much waste volume as the reference ALWRs, although about the same heat and actinide loadings. -High conversion PTRs and PTRs with DUPIC fuel cycles significantly reduce the generation of high-level waste.
SU-3	Proliferation Resistance		E			On-line refueling
S&R-1	Safety and Reliability		E			
S&R-2	CDF					
S&R-3	Mitigation					Passive Calandria heat sink for LOCA
E-1	Life-cycle cost					-The 13.5% enrichment and & other features of the High Conversion PTR lead to higher costs -The NG CANDU builds on the current low O&M costs at Darlington (low Fuel Costs).
E-2	Financial Risk					-Other concepts -NG CANDU

Appendix W5

Supercritical Nuclear Water Reactor Systems Concept Set Report

December 2002

CONTENTS

FIGURES

TABLES

ABSTRACT

Supercritical nuclear power reactors are a class of high temperature, high thermal efficiency water-cooled reactors that operate above the thermodynamic critical point of water (374°C, 221 bar) (705°F, 3208 psia). These nuclear steam supply systems may have a thermal or fast neutron spectrum depending upon the specific core design. Both light water and heavy water moderation and cylindrical as well as spherical fuel elements (i.e., pebble bed) have been proposed.

The unique thermo-physical properties of supercritical water combined with the higher, proposed system temperatures and resultant thermal efficiencies make the supercritical water thermal reactor systems good candidates for further assessment. The key issues that will emerge for determining the relative ranking of these systems are materials compatibility, reactor safety, and fuel cycle performance. The latter issues are related to the need for active engineered safety features and innovative fuel cycle measures for proliferation resistance.

The Generation IV Technical Working Group for Water Reactors feels that this concept should be further considered as a Generation IV nuclear energy system concept. However, to make such a system technologically feasible, advances are required in high-temperature materials to improve corrosion and wear resistance (fuel cladding, reactor core structural materials, and pressure boundary structural materials), in reactor core design to improve fuel-cycle versatility with these advanced materials, as well as in the reactor core, primary coolant system, and emergency core cooling designs to insure passive safety and stability.

W5.1. INTRODUCTION

W5.1-a. Background and Motivation for the Concept

Supercritical light water reactors operate above the critical temperature and pressure for water (374°C, 221 bar) (705°F, 3208 psia). Key advantages to the concept are derived from the higher temperature during heat addition to the power cycle.

- Significant increases in thermal efficiency can be achieved relative to current generation light water reactors (LWRs). Estimated efficiencies are in the range of 40-45% [Oka 2000, Bushby et al. 2000, Kitoh et al. 2001] compared to 32-34% for state of the art LWRs (Figure 1). The efficiencies shown in Figure 1 are net (MWe/MWt) efficiencies as reported in the literature.

- A higher heat transfer rate per unit mass flow results from the large specific heat above the critical point (Figures 2 and 3). This leads to: a) a reduction in the reactor coolant pumping power, b) higher fuel cladding-to-coolant heat transfer coefficients, and c) reduced frictional losses due to lower steam mass flow rates.

- A lower coolant mass inventory results from the reduced coolant density (Figure 4) as well as a lower reactor coolant system heat content (Figure 5). This results in lower containment loading from a design basis loss of coolant accident (LOCA).

- No departure from nucleate boiling (DNB or dryout) exists due to lack of a second phase, [Oka 2000, Bushby et al. 2000] thereby eliminating heat transfer discontinuities within the reactor core. However, an excessive increase in heat flux and/or decrease in coolant flow will cause heat transfer deterioration in supercritical water-cooled reactors [Tanaka et al. 1996]. The deterioration phenomena and heat transfer coefficient can be predicted by numerical simulation because supercritical water is a single-phase fluid [Koshizuka et al 1995].

- Because the coolant does not undergo a change of phase, the need for steam dryers, steam separators, and re-circulation pumps, as well as steam generators, is eliminated.

- The high coolant outlet temperatures achievable with supercritical water-cooled reactors may allow these plants to be used to produce hydrogen.

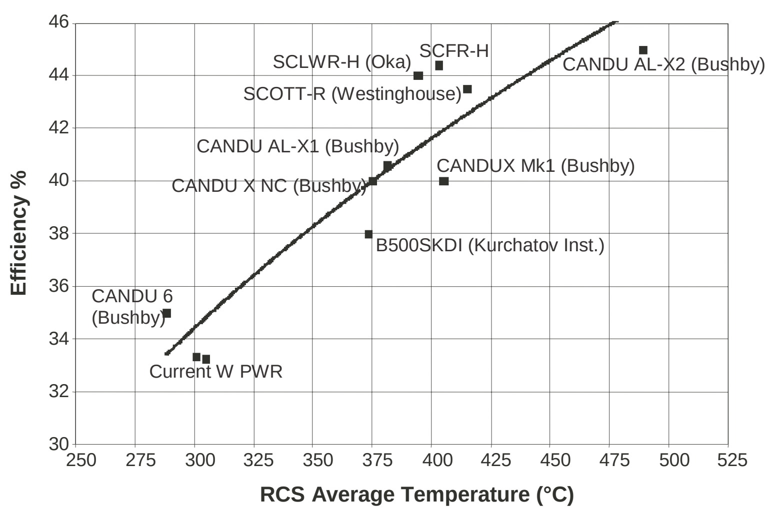


Figure 1. Effect of operating temperature on net efficiency.

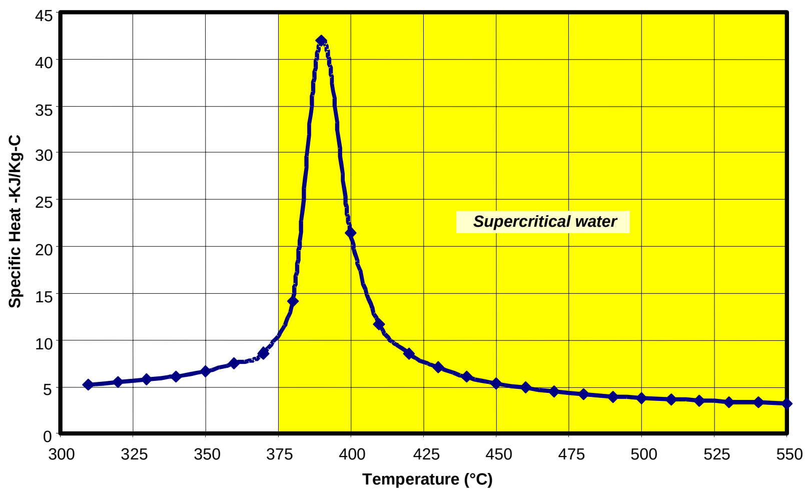


Figure 2. Enthalpy of light water between 300 and 550°C.

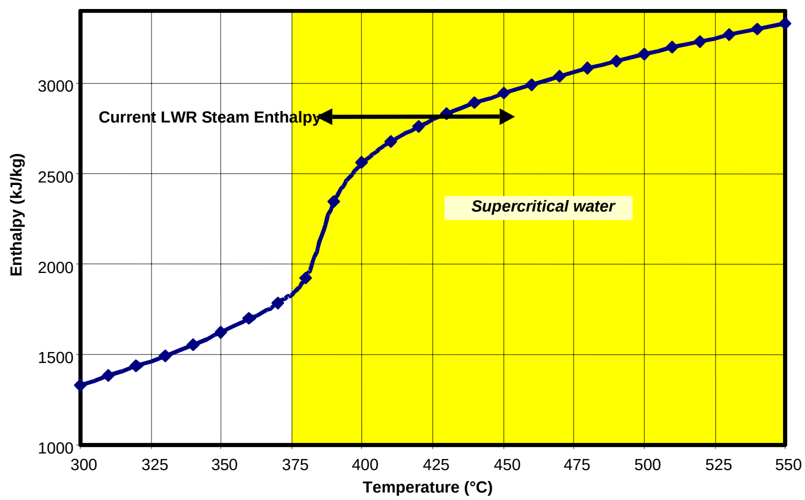


Figure 3. Specific heat of light water from 300 to 550°C at 250 bar.

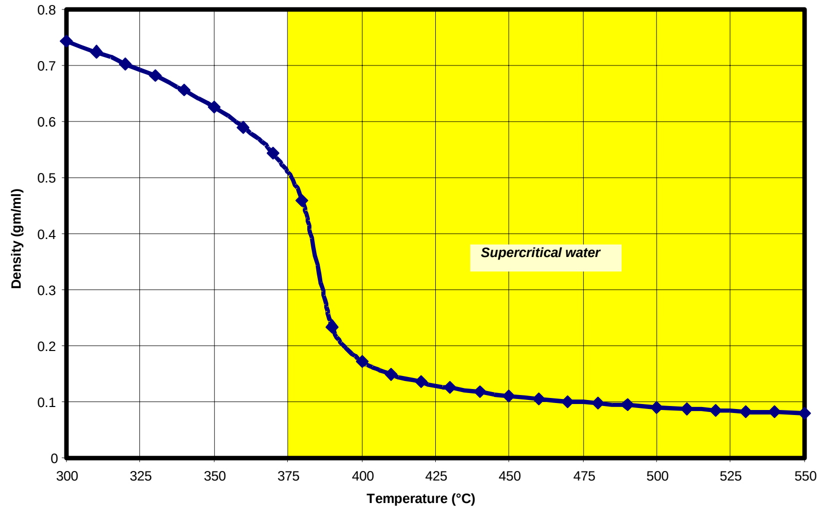


Figure 4. Density of light water from 300 to 550°C at 250 bar.

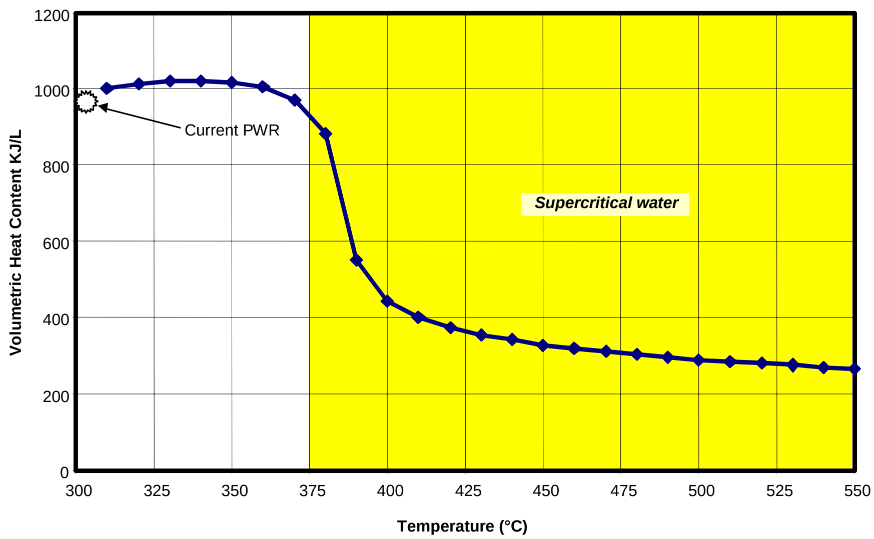


Figure 5. Volumetric heat content from 300 to 550°C at 25 MPa.

W5.1-b. National and International Interest

Interest in supercritical thermal systems extends back to the 1950's and 60's and these systems have been studied extensively in several countries. Since 1980, the major research on supercritical water-cooled reactors has mainly been in Japan and Canada. Kitoh and coworkers [2001] have considered supercritical water in fast reactor designs. Bushby and coworkers [2000] have suggested advanced heavy-water reactor designs cooled by supercritical water in pressure tubes. Oka and Koshikuza [2000] have also suggested supercritical water for various thermal spectrum LWR designs. These suggested modifications to the reactor-coolant operating parameters could result in substantial improvements (potentially up to a 25% increase) in the plant thermal efficiencies compared to current power plant systems. These possible improvements are due to the higher coolant temperature as well as plant simplifications due to changes in the plant design due to reduction in the needed components; e.g., steam generators, steam dryers, or steam separators. For indirect cycles, passive safety may also be enhanced through the improvement in the natural circulation due to the large density changes as well as large changes in heat capacity and thermal expansion in the supercritical region. Oka [2000] provides a useful review of the history of this concept. An overview of historical work on these concepts can be found in Table 1. Advanced Boiling Water Reactor (ABWR) data are added for comparison and reference. The following countries and organizations have contributed studies of various levels of detail: Brazil, Canada, European Commission, Japan, Russia, and the U. S. In the USA, Westinghouse, Babcock & Wilcox, and GE Hanford have performed studies considering the benefits of a supercritical water reactor system.

Table 2 lists the concepts that were either identified by the Generation IV Technical Working Group on Advanced Water Cooled Reactor Systems or submitted to the DOE in response to their request for information. Concept submittal W6 from the Atomic Energy of Canada (AECL) included four somewhat different and separate concepts that are listed on Table 2 separately.

Some of the current US NERI-funded research programs on supercritical water-cooled reactor design include:

1. "Supercritical Water Nuclear Steam Supply System: Innovations in Materials Neutronics and Thermal-Hydraulics" (M. Corradini, University of Wisconsin/Madison - #2001-091). The scope of this research includes investigation of ion implantation surface modification techniques to improve

materials compatibility at supercritical conditions; fuel cycle studies of enrichment needs, refueling intervals, recycling, and conversion/breeding; studies of proliferation resistance and sustainability at low coolant density; and thermal hydraulic studies to investigate heat transfer and flow stability.

2. "Feasibility Study of Supercritical Light Water Cooled Fast Reactors for Actinide Burning and Electric Power" (MacDonald 2001). This project focuses on fast spectrum systems. However, many aspects of the investigation will support thermal systems development as well, including fuel cladding and structural material corrosion and stress corrosion cracking studies (University of Michigan and the Massachusetts Institute of Technology) and various plant engineering and reactor safety analysis studies including investigations of system stability, anticipated transients without scram, LOCAs (including containment issues), startup/shutdown operation, and loss of flow accidents (Westinghouse and INEEL).

3. "Radiation-Induced Chemistry in High Temperature and Pressure Water and its Role in Corrosion" (D. Bartels, ANL, in collaboration with AECL-99-0276).

In Japan, the following programs are active:

1. The University of Tokyo, (funded from the Japan Society for Promotion of Science, 1998-2002). The research subjects are:

   Pulse radiolysis and water chemistry

   Heat transfer deterioration

   Materials.

2. Toshiba, Hitachi, The University of Tokyo, Kyusyu University, (funded from the Japanese NERI program of Institute Applied Energy, the budget is from the Ministry of Economy and Trade, 2000–2004). The research subjects are:

   Reactor and plant system studies (Toshiba, Hitachi, University of Tokyo)

   Thermal hydraulics and experimental heat transfer studies (Toshiba, Kyusyu University, University of Tokyo)

   Materials and water chemistry studies (Hitachi, University of Tokyo).

1. The University of Tokyo, 1989-. In-house study of supercritical water-cooled reactor concepts with financial support from TEPCO.

2. JNC is evaluating supercritical water-cooled reactor concepts under its Feasibility Study for deployment of a fast breeder reactor and related fuel cycle.

The R&D plans of the European Commission for a High Performance LWR operating in the supercritical regime are also of note and are outlined in Huesner et al. [2000].

Table 1. Summary of historical supercritical-water-cooled reactor concepts.

Country	Organization	Concept Name	Time	Moderator	Rating MWe	Outlet Temp °C	Pressure Bar	Net Efficiency %	Comments
USA	GE	ABWR		H_2O	1371	287	72	34.9	• For reference. Operates at saturated conditions
USA	Westinghouse	SCR	1957	H_2O	21.2	538	276	30.3	• Low efficiency due to indirect cycle • Avoided crossing critical temperature
USA	GE Hanford	Heavy Water SCR	1959	D_2O	~120	621	379	~40	• Inconel-X Clad
USA	Westinghouse	SCOTT-R	1962	Graphite	1000	566	241	43.5	• Multi-pass, pressure tube
USA	Westinghouse	SCPWR	1966	H_2O	800	371	241	33.3	• Indirect cycle with a once-through SG
Japan	Univ. of Tokyo	SCLWR	1992	H_2O	1700	508	250	44	• Once-through, direct cycle
Russia	Kurchatov Institute	B500SKDI	1992	H_2O	515	381.1/378.8 (BOC/EOC)	235	38	• Integral steam generators-steam pressure= 100 bar
Canada	AECL	CANDU-X	1998						
		CANDU-X Mark1		D_2O	910	430	250	41	• Indirect cycle, forced circulation
		CANDU-X NC		D_2O	370	400	250	40	• Indirect cycle, natural circulation

Table 1 (continued.)

Country	Organization	Concept Name	Time	Moderator	Rating MWe	Outlet Temp °C	Pressure Bar	Net Efficiency %	Comments
Canada	AECL	CANDU-ALX1	1998	D_2O	950	450	250	40.6	• Dual-cycle-supercritical reactor feeds VHP turbine. VHP turbine exhaust feeds SG with traditional indirect cycle
		CANDU-ALX2		D_2O	1143	625	250	45	• Dual-cycle-supercritical reactor feeds VHP turbine. Exhaust feeds heat exchange regenerator and SG w indirect traditional cycle
European Comm.	FRG Karlsruhe, et. al.	HPLWR	2000	H_2O	##	##	##	##	## Design parameters for the High Performance Light Water Reactor (HPLWR) are not yet defined
Brazil	Federal Unv. Of Rio Grande do Sul	Small Modular Fluidized Bed LWR	1996	H_20	~0.4/ Module	416	250	~40	• Sefidvash, 1996
USA	PNNL, RPI	Pebble Bed BWR	2000	H_20	600	540	240	44.8	• Tsiklauri, 2001

Table 2. Proposed Generation IV supercritical-water-cooled reactor concepts.

Concept #	Organization	Concept Name	Moderator	Rating MWe	Outlet Temp, °C	Pressure, Bar	Net Efficiency %	Comments
W21	Univ. of Tokyo	Thermal spectrum supercritical-water-cooled reactor (SCLWR)	H_2O	1700	508	250	44	Once-through, direct cycle
TWG1	Water TWG	Fast spectrum supercritical-water-cooled reactor	H_2O	1500/ Mono-lithic	Varied	Varied	38-45	Once-through, direct cycle Can burn actinides
W6-1	AECL	CANDU-X Mark1	D_2O	910	430	250	41	Indirect cycle, forced circulation
W6-2	AECL	CANDU-X NC	D_2O	370	400	250	40	Indirect cycle, natural circulation
W6-3	AECL	CANDU- ALX1	D_2O	950	450	250	40.6	Dual-cycle-supercritical reactor feeds VHP turbine. VHP turbine exhaust feeds SG with traditional indirect cycle
W6-4	AECL	CANDU- ALX2	D_2O	1143	650	250	45	Dual-cycle-supercritical reactor feeds VHP turbine. VHP turbine exhaust feeds SG and core inlet regeneration.
W2	PNNL	Pebble Bed BWR w/supercritical steam	H2O	200	540	240	40	Fluidized bed of SiC-PyC-coated UO2 particles in supercritical steam

W5.2. CONCEPT DESCRIPTION

Supercritical fission reactors are characterized as follows.

- They operate at high temperatures and pressures (above the light water critical point of 374°C, 221 bar).
- High thermal efficiencies (up to 45%) are achievable.
- Very compact nature of the physical plant.
- Single-phase fluid with no re-circulation.
- Both direct and combined direct/indirect cycles.
- Both light and heavy water moderated concepts are proposed.

The thermal reactor concepts are summarized in Section W5.2-a, the heavy water moderated pressure tube concepts are discussed in Section W5.2-b, the fast reactors are described in Section W5.2-c, and the pebble bed reactor is discussed in Section W5.2-d. The developers of the concepts primarily wrote the concept summaries reported below. They have been edited for style and brevity. Some of their statements may not reflect the judgment of the Technical Working Group, which is reported instead in Section 3 of this appendix.

W5.2-a. Supercritical Light Water Cooled Thermal Reactors

The Japanese supercritical light water thermal reactor (SCLWR) [Oka & Koshikuza 2000] is probably the most technically developed at this point; although Kitoh and coworkers [2001] have also considered a fast reactor variation and Bushby and coworkers have considered a supercritical CANDU [Bushby et. al. April and November 2000]. The SCLWR reactor vessel is similar in design to ABWR. High-pressure (250 bar) coolant enters the vessel at 280°C. The inlet flow splits, partly to a down-comer and partly to a plenum at the top of the core to flow downward through the core in special water rods to the inlet plenum. This strategy is employed to provide good moderation at the top of the core.

The coolant is heated to 508°C and delivered to a secondary cycle which looks like a blend of LWR and supercritical fossil technology: high- intermediate- and low-pressure turbines are employed with two re-heaters as in ABWRs.

The core consists of 211 canned hexagonal shaped fuel assemblies containing:

- 258 fuel rods (Ni-alloy clad UO_2) in a triangular array
- Guide tubes for 9 control rodlets for rodded assemblies
- 30 water rods (fed by down flow from the upper plenum)
- An instrumentation tube for fuel assembly monitoring.

The core has an effective core height of 3.2 meters and operates at a power density of 144 MW/m^3, about three times that of ABWR. The high power density is attributable to (a) the large specific heat of the coolant above the critical point (see Figures 2 and 3) and (b) elimination of the critical heat flux/

burnout phenomenon since supercritical fluid cannot be driven to a phase change by high local heat flux or high power/flow ratios.

Control rods are inserted into the top of the core as for a pressurized water reactor (PWR). Because of the large density coefficients of reactivity, temperature is controlled by control rods while power is controlled by the variable feed-water pump speed. This creates a rather complex startup strategy.

The safety system design is similar to a current generation BWR. Low core flow is the main initiating signal for protective action. Key safety systems are a high-pressure auxiliary feed-water system, a low-pressure core injection system, and a reactor scram system. The auxiliary feed-water system is based on a fast acting turbine driven pump to provide rapid response for loss of flow accidents and small break LOCA's. A turbine bypass system and safety relief valves provide system overpressure protection. See Figure 6.

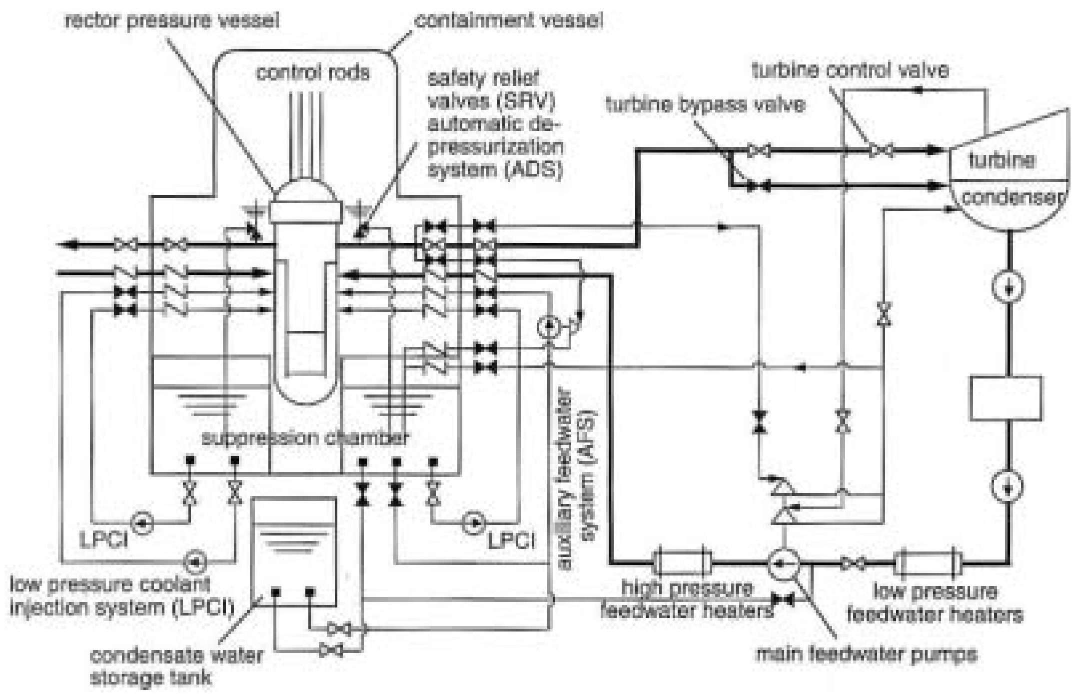
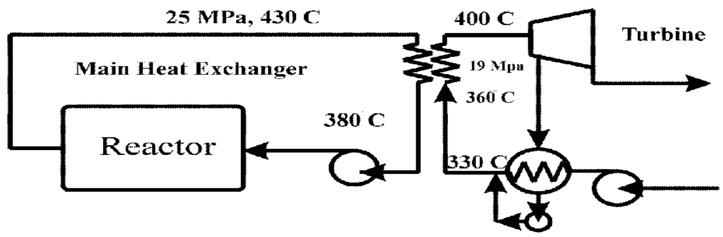


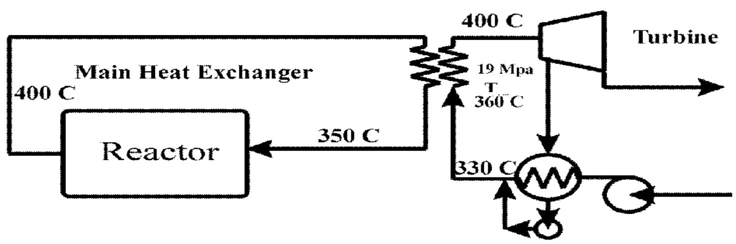
Figure 6. Supercritical light water reactor plant and safety systems.

W5.2-b Supercritical Light Water Cooled, Heavy Water Moderated Reactors

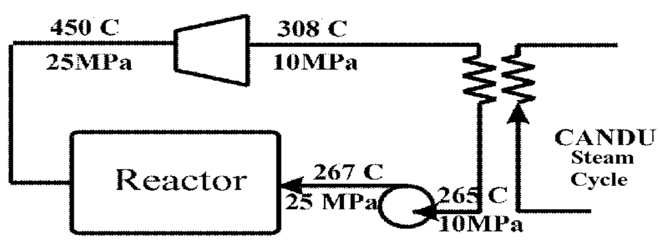
The CANDU systems [Bushby et. al. April and November 2000] appear to be at almost the same level of conceptual maturity as the SCLWR. AECL has investigated both indirect cycles with steam generators (shown in the top two sketches of Figure 7) and combined superheated /saturated steam cycles with and without reheat using very high-pressure turbines (shown in the bottom two sketches of Figure 7). They have also examined a lower power system with natural circulation on the primary side. Key system parameters are listed in Table 1. These next generation CANDU designs are based on many of the standard CANDU features including horizontal pressure tubes fueled with short fuel bundles and surrounded by a low-temperature heavy water (D_2O) moderator (on-line refueling is possible but not required in these designs).



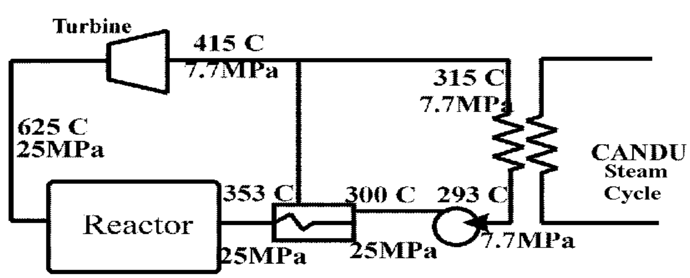
CANDU-X Mark 1



CANDU-X NC



CANDUal - X1



CANDUal - X2

Figure 7. Supercritical water-cooled CANDU cycle concepts.

The major innovations in these supercritical CANDU energy systems relevant to current CANDUs are:

- More compact core design (pressure tube spacing and fuel lattice spacing to improve overall cost and safety issues)
- Slightly enriched urania fuel in pressure tube bundles
- Higher thermal efficiency caused by higher outlet temperatures as well as higher pressures in tubes
- Enhanced passive safety systems.

Conceptual designs for four reactors have been developed, each operating at a nominal pressure of 25 Mpa. The outlet temperature for each reactor was defined by the maximum that could be achieved by a specific component. For three of the concepts (CANDU-X Mark 1, CANDU-X NC and CANDUal-X1), the temperature is based on the use of collapsible fuel cladding fabricated from alloys of Zircaloy, which sets the maximum coolant temperature at roughly 450°C.

The fourth concept, CANDUal-X2, would operate at the highest outlet temperature that can be achieved with existing supercritical water turbines (625°C). This allows for increased thermal efficiencies up to 45%. One should also note that as the design temperature rises above 500°C, hydrogen production by various technologies becomes another potential energy product. Since this temperature is beyond the operating envelope for alloys of zirconium, cladding made from other alloys must be considered to achieve the performance goals.

The salient parameters and schematic layouts for the various supercritical water-cooled CANDU concepts are presented in Table 2 and Figure 7, respectively. Many of the design choices have been made on the basis of minimizing capital and operating costs. However, it is important to note that the inherent versatility of the pressure-tube design could allow the core to be matched to the market-defined thermal cycle, turbine set and/or plant design output.

W5.2-c. Supercritical Light Water Cooled Fast Reactors

Supercritical water reactors can also be designed to operate as fast reactors. The difference between a thermal and a fast supercritical water-cooled reactor is in the lattice pitch and the use of additional moderator material. The fast spectrum reactors use a tight lattice and no additional moderator material, whereas the thermal spectrum reactors need both a loose lattice and additional moderator material in the core. Among fast reactor designs, a further distinction is whether the reactor will act as a converter or a breeder. The Japanese design [Kitoh et. al. 2001] uses mixed U-Pu oxide fuel consisting of depleted uranium and plutonium discharged from pressurized water reactors. Stainless steel is chosen as the cladding material because of its strength and corrosion resistance at high temperatures. The fuel rods are arranged in a tight triangular pitch without use of ducts around the fuel assemblies. The core arrangement consists of a central inner blanket, inner and outer seeds, a blanket between the seeds, and an outside radial blanket, surrounded by reflector shield assemblies. There is also an axial blanket. This core arrangement was adopted to accommodate the use of layers of zirconium-hydride ($ZrH_{1.7}$) between seeds and blankets to ensure a negative coolant void reactivity. The $ZrH_{1.7}$ layers are bounded by stainless steel layers and are placed in the blanket fuel assembly, one or two fuel rod rows inside from the surface to reduce the power spike in the seed. Each seed is divided into three equal-volume regions (rings) corresponding to the burnup.

Calculations [Oka and Koshikuza 2000] show that complete and partial negative void reactivity is achieved using the thin zirconium-hydride layers placed between the seed and blanket regions. Fast neutrons leaked from the seed are slowed down or thermalized in the layer due to the scattering with the hydrogen. The neutrons incoming to the blanket region have reduced energies, resulting in a higher capture to fission ratio in the blanket region. Since the effect of neutron leakage is negligible compared with absorption and production, the balance between the neutron absorption and production governs the void reactivity. The negative value arises because of the increased absorption and reduced production in the blanket regions during void conditions. The k_{eff} monotonically decreases from a coolant density of 1.0 g/cm^3 to the completely voided condition, always giving a negative void reactivity coefficient. Positive reactivity insertion during core flooding is managed by control rods, as in a BWR.

If breeding is not an objective, a simpler design can be pursued. MacDonald et al. [2001] have proposed the use of a simple, blanket-free pancake shaped core with streaming assemblies to make a fuel self-sufficient reactor that retains a hard spectrum to effectively burn plutonium and minor actinides from LWR spent fuel, while efficiently generating electricity. This is a passively safe, high leakage core that can use either fertile or fertile-free fuel, depending on whether the objective is to maximize the actinide burning or maximize plant capacity factors and minimize fuel cycle costs.

W5.2-d. Supercritical Light Water Cooled Pebble Bed Reactor

This reactor, shown in Figure 8, has unique inherent safety features due to the following [Tsiklauri 2001]:

- Ceramic coating layers are used to protect the graphite components in both air and steam at high temperatures (450-1600°C).
- The small fuel elements may be able to confine most fission products indefinitely at a temperature of 1600°C, and for several hours at temperatures up to about 2100 °C.

Pebble bed reactor fuel elements with an external coating of silicon carbide were tested out-of-pile in a high-pressure water facility (190 bar, 350°C, and PWR water chemistry) for 18 months in Russia. Spherical kernels of UO_2 (1.64 mm diameter) were coated with three layers to produce 2 mm diameter balls. The first layer consisted of pyrolytic carbon (PYC) with a 0.085 mm thickness and a density of 1 g/cm^3. The second layer consisted of dense PYC with a 0.005 mm thickness. The third layer was made of silicon carbon with a 0.08 mm thickness. The balls performed well.

The uranium loading in a 600 MWt pebble bed reactor core (1-meter radius and 2-meter height) will be about 5.1 metric tons. The fuel pebbles are loaded at the top of the reactor core and are discharged at the bottom. The discharge exposure is about 40,000 MWd/MT. The fuel residence time is about 1 year. The ^{235}U enrichment of the discharged fuel pebbles is about 2.0 weight percent.

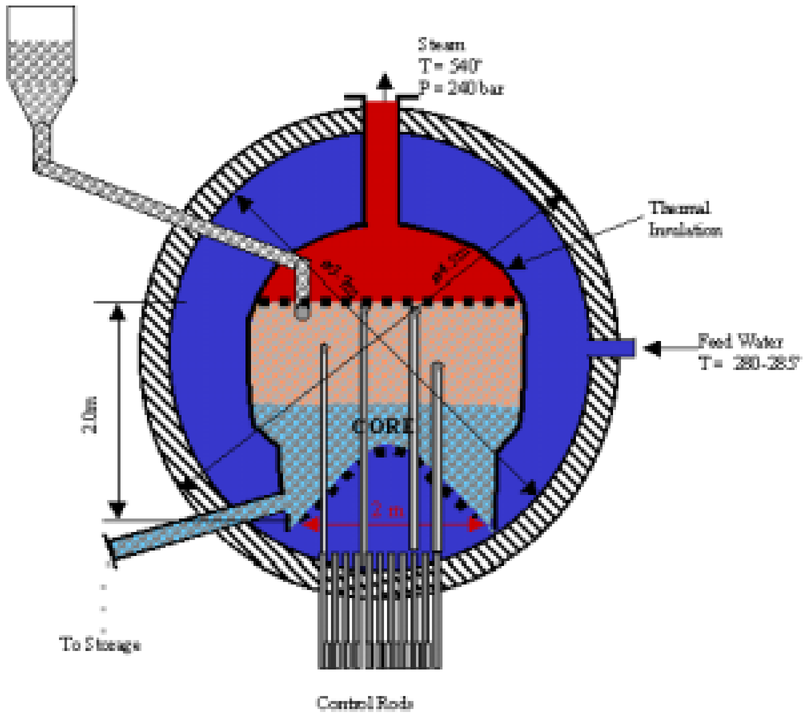


Figure 8. Pebble bed reactor with supercritical steam.

W5.3. POTENTIAL FOR CONCEPT MEETING GENERATION IV GOALS

In the following sections, the supercritical-water reactor concept set is assessed against the Generation-IV goals. The advantages and/or disadvantages of the supercritical-water reactor concept set are evaluated relative to a typical Generation III reactor (e.g., the AP-600, ABWR, and System80+ designs), which serve as the reference system. In those areas for which no appreciable differences can be identified between the supercritical-water reactor concept set and the reference, the analyzed concept is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV Criteria and Metrics by means of a label in parenthesis.

W5.3-a. Evaluation Against High Level Criteria

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

Supercritical systems have the following relative advantages with respect to Sustainability-1:

Thermal spectrum reactors:

- The high thermal efficiencies of supercritical water-cooled reactors result in efficient fuel utilization, i.e., there will be less uranium ore required per MWe-hr of electrical generation. (SU-1-1, -2, -3)

Fast spectrum reactors:

- The high thermal efficiencies result in efficient fuel utilization, i.e., there will be less ore required per MWe-hr of electrical generation. (SU-1-1, -2, -3)

- Fuel self-sufficiency is possible. (SU-1-1).

Supercritical systems have the following relative disadvantages with respect to Sustainability-1:

- None.

We conclude that overall, the high thermal efficiency and good fuel utilization is a strong advantage for supercritical water reactors in comparison to the reference ALWRs. The potential for high conversion/breeding provides an added advantage for the fast spectrum supercritical water-cooled reactors.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

Supercritical systems have the following relative advantages with respect to Sustainability-2:

- The high thermal efficiencies of supercritical water-cooled reactors result in efficient fuel utilization, i.e., there will be less high-level waste volume per MWe-hr of generation. (SU2-1, -3)

- The high thermal efficiencies also reduce thermal pollution (SU2-2)

- For the fast spectrum supercritical water-cooled reactors with breeding ratios > 1.0, the fuel utilization is substantially increased, which further reduces the high level waste volumes. (SU2-1, -3).

Supercritical systems have the following relative disadvantages with respect to Sustainability-2:

- Potentially high levels of activated corrosion products could complicate and add to decontamination, decommissioning and as low as reasonably achievable (ALARA) exposure control expenses. (SU2-2).

Overall, supercritical water reactors will minimize waste, reduce the nuclear waste stewardship burden ranging from a moderate amount (thermal spectrum reactors) to a large amount (fast spectrum reactors).

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

Supercritical systems have the following relative advantages with respect to Sustainability-3:

- None.

Supercritical systems have the following relative disadvantages with respect to Sustainability-3:

- The fast spectrum supercritical water-cooled reactors require fuel recycling. If the Purex process is used, weapons material is more readily available. If dry recycling is used, the process is much more proliferation resistant than the Purex process, and about the same as the once-through fuel cycle.

The technology could support the once-through thoria fuel cycle with proper design and implementation.

The overall impact on proliferation resistance ranges from less proliferation resistance (fast spectrum reactors) to about equivalent (thermal spectrum reactors and fast reactors with dry recycle).

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

Supercritical systems have the following relative advantages with respect to Safety and Reliability-1:

- No departure from nucleate boiling (DNB or dryout) exists due to lack of a second phase, thereby eliminating heat transfer discontinuities within the reactor core. (SR1-3)

- Elimination of many major steam-handling components (steam generators, steam separators, steam dryers) will increase reliability. (SR1-3)

- For some specific concepts there is the potential for natural circulation capability, e.g., the CANDU-X NC & Russian B500SKDI designs, see Oka [2000]. (SR1-2, -3)

Supercritical systems have the following relative disadvantages with respect to Safety and Reliability-1:

- The integrity of core materials in supercritical water presents a challenge to reliability. (SR1-3)

- The safety systems required for supercritical water-cooled reactors will generally be similar to those of the current generation in level of complexity (potential exceptions may be small integral reactors and a Pebble Bed BWR [Tsiklauri et al. 2001]). (SR1-1, -2)

- Higher pressure presents a challenge to reliability. (SR1-3).

First-of-a-kind supercritical water reactor systems are likely to have moderately lower reliability than current LWRs. Later generations should have reliability levels equivalent to the reference design.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

Supercritical systems have the following relative advantages with respect to Safety and Reliability-2:

- The reduced coolant density (Figure 4) results in a lower reactor coolant mass-inventory and reactor coolant system heat content (Figure 5), resulting in lower containment loadings during a LOCA for a given reactor power. (SR2-3)

- Initial estimates of core damage frequency (CDF) are also similar to ABWR; i.e., $< 1 \times 10^{-6}$ events/year [Oka and Koshikuza 2000]. (SR2-1, -2).

Supercritical systems have the following relative disadvantages with respect to Safety and Reliability-2:

- The safety systems required for supercritical water-cooled reactors will generally be similar to those of the current generation in level of complexity (potential exceptions may be small integral reactors and a Pebble Bed BWR [Tsiklauri et. al. 2001]). (SR1-1, -2)

A literature review yielded little data on CDF estimates. Oka and Koshikuza [2000] provide CDF estimates for the supercritical water cooled fast breeder reactor ranging from 6×10^{-7} to 7×10^{-6}/Yr.

Overall, the likelihood and degree of core damage is equivalent with respect to the reference design but significant uncertainties exist.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

Supercritical systems have the following relative advantages with respect to Safety and Reliability-3:

- The low supercritical water-cooled reactor coolant mass and energy inventory (Figure 5) should make it easier (technically feasible and economic) to achieve acceptable containment release frequencies. (SR3-1, -3)

- Generally known safety technologies are similar to ABWR. (SR3-2).

Supercritical systems have the following relative disadvantages with respect to Safety and Reliability-3:

- The safety systems required for supercritical water-cooled reactors will generally be similar to those of the current generation in level of complexity (potential exceptions may be small integral reactors and a Pebble Bed BWR [Tsiklauri et. al., 2001]). (SR3-4)

These issues are listed again since they impact both Goals 2 and 3. Core damage and large early release frequencies are not well quantified as of yet, but they are expected to be of the same order as the reference (e.g., the AP600 system). Passive safety features can be integrated into the specific reactor system design for improved performance.

Supercritical water systems are approximately equivalent to the reference ALWRs in terms of mitigation.

Economics–1. Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

Supercritical systems have the following relative advantages with respect to Economics 1 and 2:

- High thermal efficiencies result in efficient fuel utilization, i.e., there will be less burnup/MWe of generation. (EC-3, 5)

- High potential thermal efficiencies result in lower capital cost per unit energy produced. The low primary side energy content provides the potential for more compact containment designs than current LWRs; i.e., lower capital costs regardless of reactor power level. (EC-1, 2)

- A smaller reactor pressure vessel, smaller containment, and elimination of steam separation equipment will significantly reduce the capital cost. (EC-1, -2)

- The ability to size the plant design and the operating pressure and temperature allows the supercritical reactor systems to utilize “off-the-shelf” components for a large part of the power cycle (e.g., turbines or pumps) (EC-1, -2, -4, -5).

Supercritical systems have the following relative disadvantages with respect to Economics 1 and 2:

- The potential corrosion issues and high-pressure issues arising from use of supercritical water could affect total capital costs (more expensive materials) and operation and maintenance costs (ALARA and/or shorter component life cycles). (EC-1, -2, -4, -5)

- Higher pressure will require higher component cost. (EC-1, -2)

- The large variation of coolant density in the core creates the need for more complex fuel management and or reactor design in some concepts, e.g., countercurrent coolant flow, axial enrichment or poison zoning to control axial power shapes. (EC-1, -3).

Life-cycle costs for first-of-a-kind are expected to be lower than those for the reference ALWRs and substantially lower for later generations. Financial risk will be moderately higher than reference designs due to the use of high-pressure equipment (partially compensated for by a more compact plant), but later generations should enjoy an advantage relative to the reference ALWRs, if the materials issues are overcome.

W5.3-b. Strengths and Weaknesses

Strengths of the Supercritical Systems

- A high thermal efficiency is possible (40–45%).

- Potential for hydrogen generation due to high outlet temperatures.

- The lower coolant mass inventory allows for a smaller containment design.

- A smaller reactor pressure vessel and containment (than the ABWR), and simplification of the coolant transport system (elimination of steam separators, steam dryers, and re-circulation pumps) will reduce the capital costs.

- In addition, the supercritical *fast* reactor could have competitive economic advantages over thermal designs due to the tight core lattice and corresponding smaller reactor pressure vessel.

- The reduced weight of the reactor pressure vessel, internals, and fluid may yield structural design advantages.

- Supercritical systems are based on proven balance-of-plant (BOP) technology, supported by many years of experience with supercritical fossil systems at these temperatures.

- The larger enthalpy rise results in lower coolant flow rates with reduced pumping power requirements.

- There is no critical heat flux phenomena (however, a less severe deterioration of the heat transport is still possible when the power and coolant flow are not properly balanced).

Technical Issues to be Addressed

- Research is needed to understand supercritical water chemistry in the reactor environment. Potential corrosion and stress corrosion cracking issues raise materials, and reliability concerns.

- There is a need for a test database of fuel cladding and core structural materials response to supercritical water.

- The thermal-hydraulic technology, particularly for fuel bundles, is not as well developed as it is for other designs.

- The large variation of coolant density in the core creates the need for more complex fuel management and or reactor design in some concepts, e.g., countercurrent coolant flow and axial enrichment or poison zoning to control axial power shapes.

- The inherent safety and associated engineered safety features are comparable to the reference ALWRs for many concepts. This presents the challenge to enhance passive safety while not increasing the associated plant system costs.

- Use of a thermal spectrum limits the economic benefits of breeding/conversion.

- The safety technology (i.e., thermal-hydraulic and system codes) needs further development. Initial results from the Japanese studies are encouraging, but uncertainties remain.

W5.4. TECHNICAL UNCERTAINTIES

W5.4-a. Research and Development Needs

The research initiatives discussed in Section 2 address the key technical issues identified thus far for this concept. To summarize, they are:

- Characterization of corrosion and related materials phenomena for supercritical water for core and primary side materials

- Development of suitable fuel cladding and core structural materials

- Development of thermal-hydraulics/neutronics computational capabilities

- Fuel cycle studies of enrichment needs, refueling intervals, recycling and conversion/breeding technologies, proliferation resistance and sustainability

- Experimental thermal-hydraulics that examine heat transfer and flow stability

- Plant engineering and reactor safety analysis, including definition and qualification of passive and active safety features

- Additional corrosion and fission products may be transported to the turbine island due to the lack of phase separation for a direct cycle design.

W5.4-b. Institutional Issues - Licensability and Public Acceptance

No difficult licensing or public acceptance issues have been identified with this concept. A technically informed public should be receptive to high thermal efficiency plant designs, particularly for electricity generation because they tend toward:

- A reduced uranium ore ulitization

- A reduced volume of spent fuel per unit of electricity generated

- Reduced thermal pollution per unit of electrical generation

- Lower electrical power generation costs

No unusual incompatibility with the proliferation resistance and sustainability goals has been identified.

W5.4-c. Timeline for Deployment

Supercritical water-cooled reactors can be deployed within the Generation IV timeline.

W5.5. INITIAL ASSESSMENT: OBSERVATIONS AND CONCLUSIONS

The unique thermo-physical properties of supercritical water combined with the high system temperatures and improved thermal efficiencies make the supercritical water reactor systems excellent candidates for further assessment. The key issues that will emerge for determining the relative ranking of these systems appear at this point to be related to materials compatibility and associated neutronics and thermal-hydraulic issues (e.g., active vs. passive engineered safety features).

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W5.7. TOP-TIER SCREENING TABLE - SUPERCRITICAL WATER COOLED REACTORS

Summary Evaluation: __X_ Retain ____ Reject

	Goal	--	-	+	++	Comments
SU-1	Fuel Utilization					Fast spectrum: high efficiency and conversion ratio Thermal spectrum: high efficiency
SU-2	Nuclear Waste		E			Fast spectrum: high efficiency and conversion ratio Thermal spectrum: high thermal efficiency
SU-3	Proliferation Resistance					-Fast spectrum, high conversion ratio reactors -Thermal spectrum reactors
S&R-1	Safety and Reliability					First of a kind: materials, H.P. component reliability Nth of a kind
S&R-2	CDF					(+) no DNB, high specific heat (-) low mass inventory
S&R-3	Mitigation					(+) low containment loading, small volume, nat. circ. (-) not passively safe
E-1	Life-cycle cost					First of a kind: materials, H.P. component reliability Nth of a kind: high thermal eff., conversion ratio (fast)
E-2	Financial Risk					First of a kind: high pressure Nth of a kind: compact, fewer components

Appendix W6

High Conversion Water Cooled Reactors
Concept Set Report

December 2002

Appendix W6: High Conversion Water-Cooled Reactors

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FIGURES

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Appendix W6: High Conversion Water-Cooled Reactors

ABSTRACT

High conversion water-cooled reactors are designed with a fast neutron spectrum, which enhances the production of fissile material from fertile material. The designs within this concept set are water reactors with tight lattices to reduce the amount of moderation. A triangular pitch is used in most cases to make the lattice tight and in some designs additional features are introduced to reduce moderation, e.g., control rod followers, high void fractions, and heavy water. There are nine design studies (seven submittals, one of which contained three separate designs) relevant to this concept set and with two exceptions they are based on boiling water reactor (BWR) technology. Because the cores are under moderated they tend to have positive void reactivity coefficients. Hence, the cores are designed with special features such as reduced height, or the inclusion of void channels, to increase leakage and make the void coefficient negative.

The high conversion water-cooled reactors have the advantage of greatly increasing fuel utilization but do this using well-known light water reactor (LWR) technology for the nuclear steam supply system. Most of these designs would require fuel recycling to take advantage of the high conversion and this introduces the complication of changing the fuel cycle significantly relative to current advanced light water reactor (ALWR) practice in the United States.

Appendix W6: High Conversion Water-Cooled Reactors

W6.1 INTRODUCTION

The principal objective in this concept set is to advance existing light water reactor (LWR) technology by redesigning the core to optimize the conversion of fertile to fissile fuel. This is done by reducing moderation in the core resulting in a hard neutron spectrum, which leads to a high conversion ratio. Considerable effort has been invested in development of the sodium cooled fast reactor to breed fissionable ^{239}Pu from natural uranium. The concept set under high conversion water-cooled reactors is an analogous alternative technology but takes advantage of the considerable experience with LWR technology. It provides a hard neutron spectrum for efficiently converting ^{238}U (or ^{232}Th) into the fissile ^{239}Pu (or ^{233}U) with conversion ratios about 1.0.

The main benefit of a high conversion water-cooled reactor is the improvement in fuel utilization. The improved fuel utilization has a secondary benefit of reduced waste. Since the core has a hard spectrum it also has the potential to be an effective waste burner for long-lived fission products and/or minor actinides. Furthermore, if the thorium cycle is used, as proposed in one high conversion water-cooled reactor project, then the approach also has advantages for nonproliferation. As with many concept sets, mixed plutonium-uranium oxide (MOX) material can be the source of fissile material. If weapons-grade, then there are obvious proliferation resistance benefits; if reactor grade, then the benefits are primarily in fuel utilization and waste management.

Since the objective of these designs is to produce fissile material, several of the designs take into account that fuel will be reprocessed as part of the fuel cycle. In some designs the fissile fuel created would only be used within the core in which it is produced, e.g., this is envisioned for the concept using thorium as the fertile material.

Interest in this concept set in the last decade has primarily come from Japan, especially through the auspices of a Japanese consortium coordinated by Japan Advanced Energy Research Institute (JAERI). Japanese design studies have been carried out both by industry (Hitachi, Toshiba, and Mitsubishi) and research institutes [JAERI, and Japan Atomic Power Company (JAPC)]. The designs are collectively known as reduced moderator water reactors (RMWRs).

In addition to the Japanese effort, the concept is currently being studied under DOE’s NERI program by researchers at Brookhaven National Laboratory (BNL) and Purdue University. This effort was begun in FY-99 and is a three-year effort. The thorium fuel cycle is utilized in the U.S. study and weapons-grade MOX fuel is used to provide fresh fissile material.

W6.2 CONCEPT DESCRIPTION

Most concepts are similar in that they use a tight lattice based on a triangular pitch to minimize moderation and produce the fast spectrum essential to achieve a high conversion ratio. Most do this within a boiling water reactor (BWR) design but pressurized water reactors (PWRs) are also considered. Since BWRs run with a void fraction in the core, which can be increased relative to a normal BWR, they can run with reduced moderator density relative to a PWR for the same lattice dimensions. The PWRs use heavy water, with its decrease in moderating power relative to light water, to compensate and provide a harder spectrum for a given configuration. Other differences between concepts are the fuel assembly geometry and the design differences related to concerns over the void coefficient, which tends to be positive in a core with a hard (under-moderated) spectrum. The latter results in most designs using flat cores in order to increase leakage during voiding and thereby make the void coefficient negative. The features actually used by different concepts are summarized in Table W6.1, which is explained in detail below.

Table W6.1. High conversion water-cooled reactor designs.

Acronym	Principal Designer	Reactor Type	Fuel Assembly (FA) Shape	Coolant[a]	VC Strategy
HCBWR (W9)	Hitachi	ABWR-II	Square	LW	Void tubes
HCBWR-Th (TWG6)	BNL	SBWR/ABWR	Hex	LW	Th cycle
SSBWR (W19)	Hitachi	Indirect Cycle BWR; Integrated System	Hex	HW changing to LW during each fuel cycle.	--
BARS (W27)	Toshiba	ABWR	Square	LW	FA with different heights
RMWR (W24)	JAERI	ABWR	Hex	LW	Double flat core
RMWR (W24)	JAERI	ABWR	Hex	LW	Void tubes
RMWR (W24)	JAERI	ABWR	Square	LW	No blanket
ISPWR (W20)	Mitsubishi	PWR; integrated system	Hex	HW	--
PWR (W30)	Mitsubishi	PWR	Hex	HW	Seed/blanket
[a]LW = light water; HW = heavy water					

Table W6.1 provides in columns 1 and 2 the acronym used and the principal designer for several concepts employing high conversion cores. There are more variations in this concept set but these represent the ones documented for the Working Group. Within column 1 the designation in parenthesis is the reactor concept identification number. The first concept is called the high conversion BWR (HCBWR) (Mochida 2001; Yamashita and Mochida 1991). The second concept is the variant of the HCBWR that uses the thorium fuel cycle (Diamond 2001, Downar 2000; Takahashi et al. 2000a, 2000b, 2001). The third concept is the Safe and Simplified BWR (SSBWR) (Ohtsuka 2001). The next concept is the BWR with an Advanced Recycle System (BARS) (Kouji 2001). The next three are variants of the reduced moderation water reactor (RMWR) (Iwamura 2001; Okubo 2000). The last two concepts in the table are the integrated system PWR (ISPWR) ((Makihara 2001) and a loop-type PWR (Hibi 2001).

The third column in the table gives the reactor type, i.e., the nuclear steam supply system (NSSS) used. In general it is the Advanced BWR (ABWR) design that would be used, however, one concept has integrated their core with a more advanced version, ABWR-II, and one intends to use aspects of the Simplified BWR (SBWR) to improve safety.

Two of the concepts use integrated primary systems and are very different from a BWR. The Safe and Simplified BWR (SSBWR) is an indirect cycle that uses a boiling system and a steam generator to produce steam in a secondary system. It is an integrated design and the steam generator is within the reactor vessel. In the integrated system PWR (ISPWR) the steam generators are inside the vessel and natural circulation is used. The ISPWR and the SSBWR share the principal goal of improving fuel utilization through high conversion and are thus grouped within this concept set even though they also introduce features similar to those found with other integrated primary systems concepts discussed in Appendix W1.

All the designs listed in Table W6.1 use tight lattices to harden the spectrum with cognizance that tight lattices make cooling more difficult. The tight lattices use a triangular pitch in all cases except for one RMWR design (the one using a square fuel assembly), which uses a square pitch. In some designs, as indicated in column 4 in Table W6.1, a square fuel assembly is used and in some it is a hexagonal fuel assembly. The two options with respect to fuel assembly geometry are shown in Figure W6.1. The square lattice takes advantage of existing BWR technology whereas the hexagonal lattice takes advantage of the more natural geometry using a triangular pitch. Some variants of the standard BWR square fuel assembly have been tried wherein the external dimensions are increased by up to a factor of two, similar to the size of PWR fuel assembly.

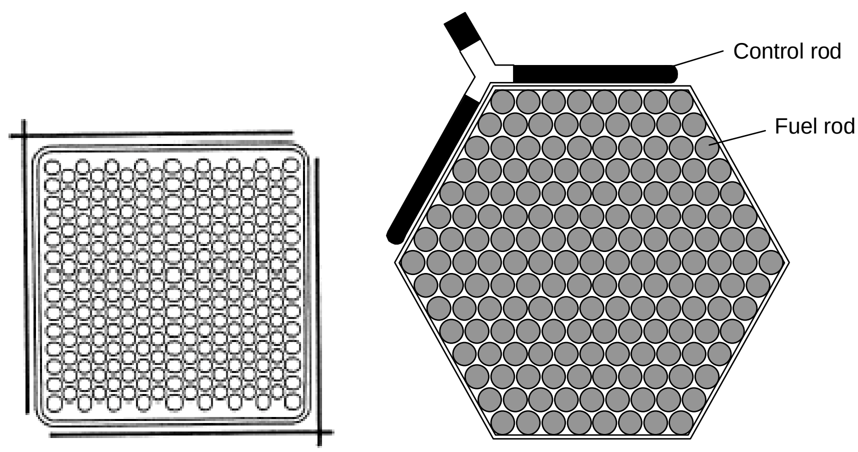

Figure W6.1. Square and hexagonal fuel assemblies with triangular pitch.

In most designs other features beside the tight lattice are necessary to either reduce moderation further and/or to improve cooling. The ISPWR and PWR use heavy water to reduce moderation whereas the BWRs take advantage of the presence of voids. The SSBWR is the only BWR that also uses heavy water as coolant to loosen the lattice and improve circulation in the core. This is feasible since the SSBWR uses an indirect cycle and the heavy water remains in a closed loop. This design also uses the spectral shift concept by diluting the heavy water with light water through the fuel cycle in order to lengthen the cycle. The use of heavy water or light water is indicated in column 5 in Table W6.1.

Another way to reduce moderation is to use a control rod follower. Figure W6.2 shows this design for the HCBWR. The water in the gap between the fuel bundles in the top part of the core contributes to moderation and the insertion of a follower (*above* the absorber region), which is an inert material, displaces the water without adding absorber. The control blade absorber region also displaces water but reduces power as well. The reactor can be operated with the follower withdrawn if it desirable to increase moderation.

One of the problems of designing a core with a fast spectrum is the tendency to have a positive void reactivity coefficient (VC) because of the under-moderation. Most of the designs use a short core (~1 m) to increase leakage and thereby make the void coefficient negative. The short core is also desirable to reduce pumping power, which is increased due to the tight lattice. To maintain the same power level with a short core requires a larger diameter core. Many other design changes have been considered to also increase the negative void coefficient and/or to allow for an increase in core height (and therefore, power). These design features are noted in column 6 of Table W6.1 and discussed below.

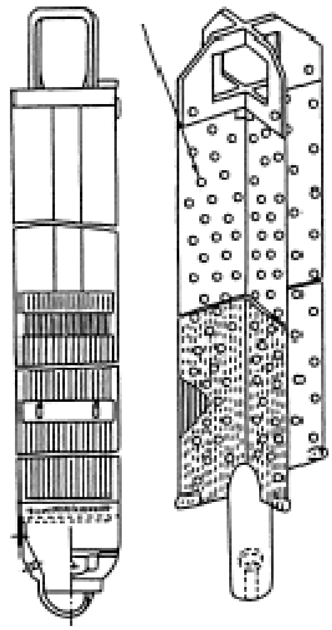

Figure W6.2. Fuel assembly and control blade assembly with follower.

One variant of the HCBWR uses void tubes within the core to increase leakage. Figures W6.3 and W6.4 are planar and elevation views of a typical HCBWR core showing the geometry that would be necessary with hexagonal fuel assemblies in a core using void tubes to control the void coefficient. For the HCBWR-Th design, the BNL/Purdue researchers have demonstrated that the physics using a thorium fuel cycle is such that no void tubes will be necessary (Downar 2000).

The BARS concept uses fuel assemblies with two different heights. This results in the long fuel assemblies having essentially no neighbors at the top of the core, thereby increasing leakage. This is shown in Figure W6.5.

There are three different RMWR designs with different objectives and they use different designs to deal with the void coefficient. The different designs are (a) to achieve a high conversion ratio (actually a breeding ratio of 1.1), (b) to obtain both a high burnup (60 GWd/t) and a two-year cycle, and (c) to simplify the design. The first design objective is obtained with a double flat core, which consists of a sandwich of two flat cores between three blankets. The second uses void tubes within the core and the third, the square pitch case, uses no blanket.

Very little has been said in the literature about control rods and what materials will be used and what their effectiveness will be. It is expected that more control rods will be required than currently used, e.g., one cruciform per BWR fuel assembly rather than the one per four bundles currently found in BWRs. If this is the case, then the mechanical design of the reactor will be more complicated than for existing reactors.

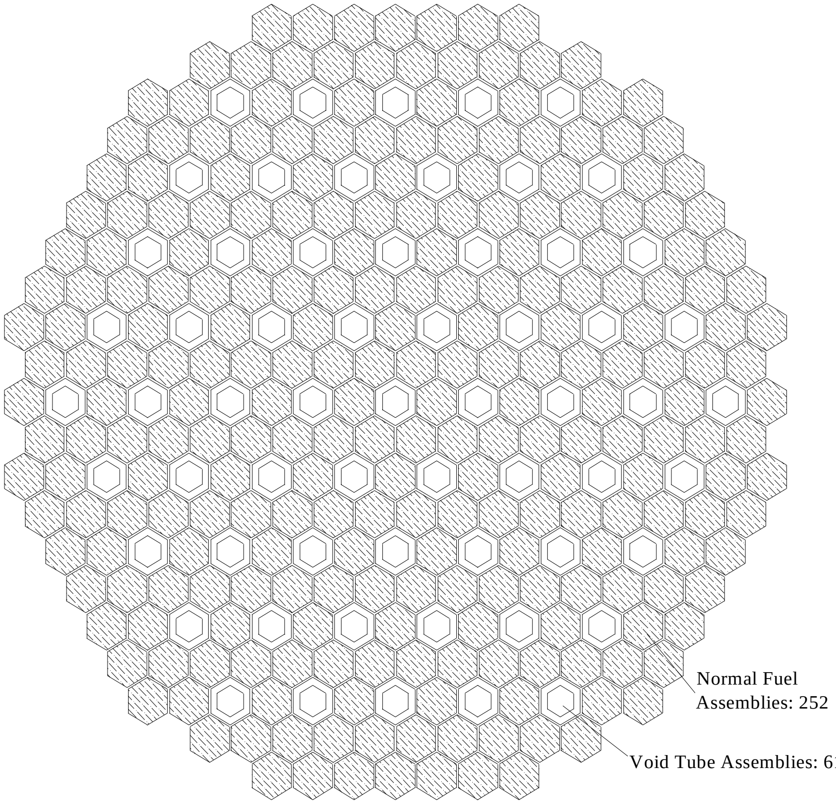

Figure W6.3. Horizontal cross section of the HCBWR.

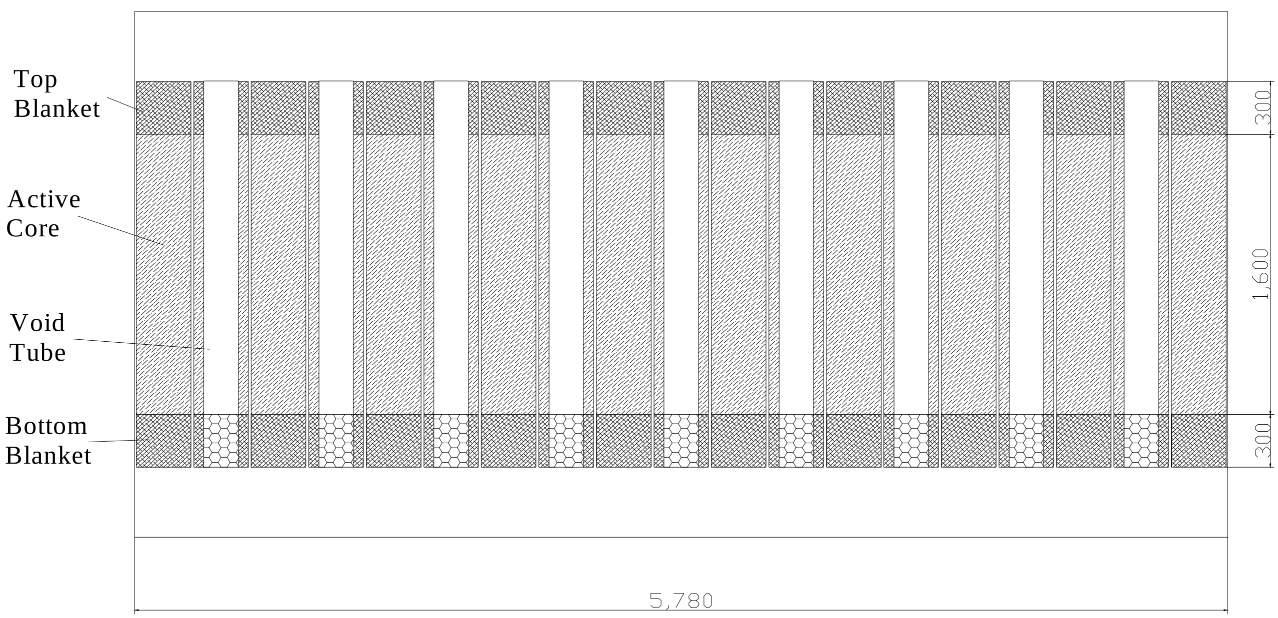

Figure W6.4. Vertical cross section of the HCBWR.

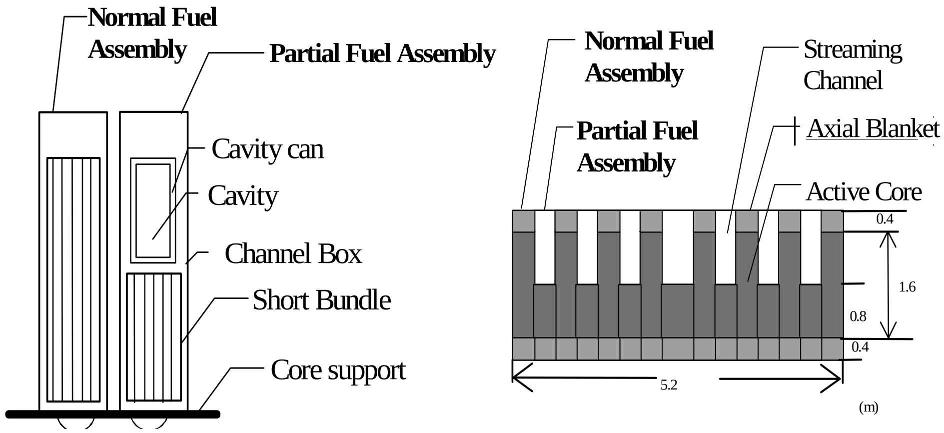

Figure W6.5. Partial assembly and core elevation with " unequal" height assemblies.

W6.3 POTENTIAL FOR CONCEPT MEETING GENERATION IV GOALS

W6.3.a. Evaluation Against High Level Criteria

In the following subsections, the High Conversion Water-Cooled Reactor concept set is assessed against the Generation IV goals. The advantages and/or disadvantages of this concept set are evaluated relative to a typical Generation III ALWR reactor with a one-through uranium fuel cycle. In those areas for which no appreciable differences can be identified between the concept set and the reference, the concept set is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and metrics by means of a label in parenthesis.

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

High conversion water-cooled reactors have the following advantages relative to the reference reactor with respect to Sustainability-1:

- Reactors with high conversion ratios significantly improve fuel utilization because they create significant amounts of fissile material. (SU1-1)
- Reactors with high conversion ratios can be used to burn the existing LWR high-level waste. (SU1-1)
- The impact on environment is reduced due to the significantly reduced ore needs. (SU1-2)

It is concluded that high conversion water-cooled reactors have exceptional advantages over the reference ALWRs. Indeed it is because of Sustainability-1 that there is an incentive for this type of design.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

High conversion water-cooled reactors have the following advantages relative to the reference reactor with respect to Sustainability-2:

- Fast reactors with high conversion factors reduce the amount of waste; particularly high-level. Therefore, the environmental impact and the stewardship burden are also reduced. (SU2-1, SU2-2, SU2-3)

- Cores with fast spectrum can be designed to burn fission products and minor actinides. (SU2-1, SU2-2,, SU2-3)

High conversion water-cooled reactors have the following disadvantage relative to the reference reactor with respect to Sustainability-2:

- The need for fuel recycling in some designs with high conversion cores introduces new industrial facilities that will have some environmental impact.

It is concluded that high conversion water-cooled reactors are much better than the reference ALWRs, primarily by virtue of the reduced quantity of waste.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

High conversion water-cooled reactors have the following advantage relative to the reference reactor with respect to Sustainability-3:

- One concept (HCBWR-Th) is specifically introduced with nonproliferation in mind. Using thorium there are nonproliferation benefits (see also Appendix W8) due to the reduction in Pu production and similarly, in the design where the fissile material comes from weapons grade Pu there are proliferation benefits because Pu is being destroyed. (Note that this assumes that the ^{233}U produced will be denatured with ^{238}U.) (SU3-3)

High conversion water-cooled reactors have the following disadvantage relative to the reference reactor with respect to Sustainability-3:

- For those cycles using the PUREX process the proliferation potential is increased. However, note that the dry recycling used in the HCBWR and BARS designs results in no change in proliferation potential and dry recycling could be used for any of the designs. (SU3-1)

It is concluded that high conversion water-cooled reactors have a disadvantage compared with the reference ALWRs when wet recycling (PUREX) is used. If dry recycling is used, then these concepts may be essentially equivalent to the ALWR fuel cycle in proliferation resistance. The nonproliferation advantages in using the thorium cycle are not unique to high conversion water-cooled reactors and could be applied to other concept sets.

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

High conversion water-cooled reactors have the following disadvantages relative to the reference reactor with respect to Safety and Reliability-1:

- The hard spectrum will introduce material problems different from that found in the ALWRs and these might reduce reliability. A number of the concepts propose the use of stainless steel for cladding the fuel and new materials may be needed for control rod absorbers. There is less experience with these materials than with the alloys currently used in LWRs and the early experience with the use of stainless steel cladding in BWRs was not good. (SR1-3)

- The pressure vessel fluence may be increased as a result of the increase in fast neutrons in the core. (SR1-3)

- In those designs where heavy water is introduced the worker exposures may increase due to the additional tritium that will be produced. It should be noted that the CANDU experience shows that worker exposure at heavy water moderated and cooled plants can be reduced to acceptable levels. (SR1-1)

- The need for more control rod drives may complicate the design and reduce reliability. (SR1-3)

- The spent fuel recycling and MOX fuel fabrication may increase worker exposures. (SR1-1)

It is concluded that worker exposures may be higher than expected with a once-through fuel cycle in ALWRs. In addition, the plant reliability may initially be less because of the use of stainless steel fuel cladding. Although, the fuel being used in high conversion water-cooled reactors is oxide and there are no significant changes in the nuclear steam supply system designs, the hard spectrum may induce stress corrosion cracking in the fuel cladding and reactor internal components.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

High conversion water-cooled reactors have the following advantage relative to the reference reactor with respect to Safety and Reliability-2:

- In a high conversion water-cooled reactor core there is a relatively low reactivity swing during the fuel cycle and this minimizes any reactivity-initiated accident. (SR2-1)

High conversion water-cooled reactors have the following disadvantages relative to the reference reactor with respect to Safety and Reliability-2:

- The design has to have enhanced neutron leakage to overcome the potential for a positive void coefficient. (SR2-1)

- The design must account for the additional friction in the coolant channels and the smaller coolant volume due to the tight lattice. Re-flood as well as normal operation must be considered. (SR2-1)

- An aspect of BWR safety that will need to be addressed is stability. Differences in core design will result in differences in stability relative to existing cores. (SR2-1)

- Although some thermal-hydraulic experiments have been carried out to understand heat removal, additional testing and analytical development will be needed before safety analyses can be confidently done. (SR2-2)

It is concluded that there are safety considerations as a result of the high conversion water-cooled reactor core design although all of these should be capable of (and must be) dealt with in the final design. Beyond the core, safety considerations in the NSSS are similar to current ALWR design. Those reactors incorporating ABWR-II or SBWR technology may have improved safety relative to an ALWR.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

It is concluded that because high conversion water-cooled reactors have similar nuclear steam supply systems and similar safety systems to existing ALWRs that they should be equivalent, i.e., they should have the same highly robust mitigation features or fission product barriers, and the same low levels of risk to individuals and society.

Economics–1. Generation IV nuclear energy systems will have a clear life cycle cost advantage over other energy sources.

High conversion water-cooled reactors have the following advantage relative to the reference reactor with respect to Economics-1:

- These cores are designed to have long fuel cycles and high plant capacity factors.

High conversion water-cooled reactors have the following disadvantages relative to the reference reactor with respect to Economics-1:

- The fuel costs associated with the recycle of plutonium back into an LWR using wet recycling (PUREX) have traditionally been somewhat higher than the fuel costs for the once-through all-uranium fuel cycle. (EC-3)

- The fuel costs associated with dry recycle of the fissile material will probably be somewhat higher than the costs associated with wet recycling. (EC-3)

- The SSBWR has a long cycle length and therefore a high capacity factor. However, it has high fuel costs because of the low power density. (EC-3)

- The SSBWR dilutes heavy water during operation and this leads to a financial penalty. (EC-3)

- Since there are currently no recycling facilities in the United States, the use of systems that require (either wet or dry) recycling will have higher costs than an ALWR using a once-through cycle.

It is concluded that in general, with the exception of the SSBWR, the operating costs for a reactor with a high conversion core should be somewhat higher than those for an ALWR due to the operational costs of the spent fuel recycling facility and the additional MOX fuel fabrication costs. An additional penalty is the capital cost of the recycling facilities. The fuel cycle costs associated with the SSBWR will be significantly higher than the costs for an all-uranium once-through fuel cycle.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

High conversion water-cooled reactors have the following disadvantages relative to the reference reactor with respect to Economics-2:

- The use of heavy water in the ISPWR, PWR and SSBWR designs increases capital costs. (EC-1)
- The requirement for additional control rods may lead to higher costs. (EC-1)

It is concluded that capital costs should not be strongly influenced by the differences in core design to achieve high conversion with the exception of the need for additional control rods. Those designs using heavy water also have a cost penalty due to the expense of the heavy water.

W6.3.b. Summary of the Strengths and Weaknesses

Strengths of the High Conversion Water-cooled Reactor Concepts

- Utilizes known water reactor technology for everything but the core; this helps stabilize costs and risk due to potential accidents
- Utilizes a hard spectrum to produce a high conversion ratio to enhance fuel utilization
- Reduces waste due to the high fuel utilization
- Has the potential to produce long fuel cycles.

Weaknesses of the High Conversion Water-cooled Reactor Concepts

- Requires design features to mitigate tendency to have positive void coefficient
- Additional control rods and control rod drives complicates the design
- Lower water volume in core requires special consideration of cooling requirements
- The harder spectrum in the core may lead to new radiation damage problems
- Some designs utilize heavy water which increases costs
- Some designs include recycling and this would increase proliferation concerns
- Recycling necessitates the development of that capability in the United States.

W6.4 TECHNICAL UNCERTAINTIES

W6.4.a. Research Needs

The research and development (R&D) done to date on this concept set has primarily been in Japan. This includes both experimental and analytical work in neutronics and thermal-hydraulics of the core design. The experimental work in neutronics will be done in the Thermal Critical Assembly (TCA).

Planning has begun, and experiments using RMWR fuel are expected to begin in 2004. Critical heat flux and re-flood experiments are already underway. Obviously there is a need for a continuation of all of this R&D work.

The recent U.S. R&D for high conversion water-cooled reactors has been primarily the NERI project at BNL. This project, to be completed within the next year, will develop a reactor neutronics and thermal-hydraulics design based on a thorium cycle but much of the research needs for this project would be needed for any high conversion water-cooled reactor core design. Critical experiments to validate the neutronics design, thermal-hydraulic experiments on tight lattices to obtain constitutive relations, and stability tests are some examples of additional needed research. Another broad area of R&D relates to fuel behavior under normal and abnormal conditions. It might also be of interest to design control rods with materials such that an increased number of control rods, relative to existing BWRs, might not be required.

There is also a need to determine which of the many design approaches now being considered will be most effective.

W6.4.b. Institutional Issues – Licenseability and Public Acceptance

There are several new licensing issues that will have to be addressed with this concept set. The safety considerations discussed above, namely, the void coefficient and adequate cooling, are one set of issues. The use of wet or dry recycling to recycle the fissile materials will be another set of issues. And another set of issues comes from the different variants proposed for use with high conversion cores. However, some of these matters, e.g., integrated system designs or the use of thorium fuel, are generic to many concept sets.

Assuming that the public becomes more comfortable with nuclear energy in the years to come, it would seem that the introduction of high conversion water-cooled reactors and fissile material recycle should not introduce an adverse reaction, and indeed its fuel sustainability characteristics should make nuclear energy more attractive.

W6.4.c. Timeline for Deployment

It can be assumed that the timeline for deployment in the United States would be similar to that expected in Japan where one of the most important incentives is the conservation of uranium resources. This assumption is based on international cooperation and a willingness in the United States to endorse recycling. In Japan the RMWR is expected to be introduced in stages with each stage having a higher conversion ratio. By about 2015, they expect to introduce a version with a conversion ratio of 0.98. Although this would be done by changing the ABWR design as little as possible, it is not clear to the Working Group that this could be achieved. By 2050, they expect to have a version with a conversion ratio of 1.13. Note that by 2050 they expect to have a liquid metal fast breeder reactor in service as well.

W6.5 INITIAL ASSESSMENT: OBSERVATIONS AND CONCLUSIONS

The concept set “High Conversion Water-Cooled Reactors” includes several designs that can greatly increase fuel utilization and reduce waste and weapons materials, while at the same time employing proven water reactor technology.

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W6.7. TOP-TIER SCREENING TABLE - HIGH CONVERSION WATER-COOLED REACTOR CONCEPTS

Summary Evaluation: __X__ Retain ____ Reject

Goal		--	-	+	++	Comments
SU1	Fuel Utilization				■	This is principal reason for this concept.
SU2	Nuclear Waste				■	Additional benefit from high conversion.
SU3	Proliferation Resistance		■	■		-Wet recycling is not as proliferation resistant as the reference ALWR once-through fuel cycle -Dry recycling and thorium fuel cycles are equivalent or better than the reference ALWR once-through fuel cycle.
S&R1	Safety and Reliability		■			The spent fuel recycling and MOX fuel fabrication may increase worker exposures. In addition, the plant reliability may initially be less because of the use of stainless steel fuel cladding.
S&R2	CDF		■	■		There are number of uncertainties including the potential for a positive void coefficient, the difficulty in reflooding a tight lattice core, BWR stability, and material damage due to the hard spectrum

S&R3	Mitigation		F			
E1	Life-cycle cost					-Concepts with fissile recycle would have somewhat higher fuel costs due to the spent fuel recycling and irradiated material fabrication. These concepts would also have capital costs associated with the recycle fuel fabrication facilities. -Concepts using heavy water dilution would have a significant economic penalty.
E2	Financial Risk		F			-Concepts that use light water coolant will have the same plant capital costs as the ALWRs. -Concepts using heavy water have higher capital costs

Appendix W7

Pebble Fuel Reactors Concept Set Report

December 2002

CONTENTS

FIGURES

TABLES

Appendix W7: Pebble-Fuel Reactors

ABSTRACT

Three reactor concepts were submitted in response to the DOE Request for Information that are characterized by the use of a fluidized-bed core with spherical fuel particles with either ceramic (TRISO) or metallic (zirconium alloy) cladding. Two concepts are based on a direct cycle heat-transport scheme, one on an indirect cycle. The coolant/moderator is light water, either pressurized, boiling, or supercritical. The primary-coolant mode of circulation is always forced. All the proposed concepts are thermal reactors and make use of low-enrichment-uranium oxide-fuel.

The emphasis in this class of reactors is on passive safety (i.e., all concepts feature passive shutdown and passive decay heat removal capabilities) and reduced fuel temperature operation because of the large heat transfer area available for removing the nuclear heat.

However, issues/concerns were identified for this reactor concept. These include: fuel reliability, fuel/coolant interaction, and fuel fabricability.

Appendix W7: Pebble-Fuel Reactors

W7.1. INTRODUCTION

W7.1-a. Background and Motivation for the Concept

The light-water-cooled Pebble Fuel Reactor (PFR) concept can be viewed as combining the attractive characteristics of the high-temperature gas-cooled reactor (e.g., good retention of the fission products at high temperature, passive decay heat removal, passive shutdown) with the traditional light water reactor (LWR) technology. The PFR concepts submitted to the Generation-IV water-reactor evaluation committee are characterized by the use of relatively large spherical fuel particles (outside diameter in the 2–10 mm range) with either a ceramic or metallic cladding. The particles are kept in suspension in the core by the water coolant flow as a fluidized bed. If a loss-of-flow or loss-of-coolant accident occurs, the fuel particles fall into a sub-critical configuration that automatically shuts down the reactor. Moreover, because of the large surface-to-volume ratio, the fuel normally operates at relatively low temperatures and the decay can be transported out by radiation and conduction even if the coolant is lost.

W7.1-b. National and International Interest

The concept of a pebble-bed light-water-cooled reactor has raised some interest worldwide in the past two decades. A highly-modular pebble-bed pressurized water reactor (PWR) with Zircaloy-clad UO_2 fuel was first proposed in 1985 by Prof. Sefidvash of the Federal University of Rio Grande do Sul in Brazil (Sefidvash 1985).

Russian scientists (Artamkin 1986; Legchilin 1987; Ponomarev-Stepnoy et al. 1999; Filippov and Bogoiavlensky 2001) published studies on pebble-bed gas-cooled and water-cooled reactors. Japanese scientists (Mizuno et al. 1986 and 1990) also reviewed this concept. In 1989, the Oak Ridge National Laboratory prepared a report for DOE (Forsberg 1989) on this reactor.

Corrosion experiments were conducted in Germany with TRISO fuel particles in steam between 600°C and 1400°C for 24 hours with natural convection conditions, and no mass loss was detected (Hurtado et al. 1992). (The word TRISO is used to describe a generic category of gas or water cooled reactor fuel that has a spherical uranium or thorium oxide or carbide or oxy-carbide kernel covered with layers of pyrolytic carbon and silicon or zirconium carbide.) Similarly, in Russia spherical fuel elements with an external coating of silicon carbon were exposed to high-pressure water (190 bar, 350°C and PWR water chemistry) for 18 months (Filippov and Bogoiavlensky 2001; Ponomarev-Stepnoy et al. 1999). A three-layer coating was used around the UO_2 kernels (1.64 mm diameter). The first layer consisted of porous pyrolytic carbon (PyC) of 85 μm thickness; the second layer consisted of dense PyC of 50 μm thickness, and the third layer was made of silicon carbon of 80 μm thickness. All samples maintained their integrity, and the mass loss was practically negligible. These TRISO fuel particles were also tested in a steam facility (100 bar, 550°C) for 15 months, and it was found that the mass loss was less than 1%. The steam temperature was gradually increased to 950°C. The testing time was decreased from 14 days to 1 day. In these experiments, mimicking the conditions of hypothetical severe accidents, the mass loss was not substantial, and all the spheres maintained their integrity. No references for either the German or the Russian studies were provided by the proponents of the PFR concepts.

A summary of the general characteristics of the three PFR concepts submitted to the Generation-IV water-reactor evaluation committee is reported in Table 1, where some additional references for one of the concepts are also provided. No references could be found for the concept proposed by the Pacific Northwest National Laboratory (PNNL), except for the Russian references. A more detailed description of the three concepts is provided in the next section.

Table 1. Summary of integrated primary-system concepts submitted to DOE for the Generation-IV Program.

Gen-IV Designation	Proponent	Size	Coolant State	Mode of Circulation	Cladding	References
W1	Tsiklauri (PNNL, USA)	200 MWe	Boiling (7.0MPa)	Direct	TRISO	Filippov and Bogoiavlensky 2001, Ponomarev-Stepnoy et al. 1999
W2	Tsiklauri (PNNL, USA)	240 MWe	Supercritical (24MPa)	Direct	TRISO	N/A
W4	Sefidvash (UFRGS, Brazil)	1 MWe per assembly	Pressurized (15MPa)	Indirect	Metallic Zr	Sefidvash 1985, 1995, and 1996

W7.2. CONCEPT DESCRIPTION

Two subgroups can be identified within the PFR concept set:

1. Concepts with TRISO particle fuel

2. Concepts with zirconium-clad fuel.

A brief description of these two subgroups is presented in Sections W7.2-a and b below, respectively. However, this categorization will not be used for evaluation of the potential for meeting the Generation-IV goals (see Section W7.3), i.e., the different PFRs will be evaluated together. The developers of the concepts primarily wrote the concept summaries reported below. They have been edited for style and brevity. Some of their statements may not reflect the judgment of the Technical Working Group, which is reported instead in Section 3 of this appendix.

W7.2-a. Pebble Fuel Reactors with TRISO Fuel (W1, W2)

These are direct-cycle reactors with a fluidized-bed core made of several million TRISO coated fuel particles.

W1 – Pebble Bed BWR

The fuel elements are small pebbles (between 2 and 10 mm diameter) consisting of low-enrichment UO_2 kernels coated with 3 layers. The inner layer is made of porous pyrolytic carbon (PyC) called the buffer layer, providing room for gaseous fission product accumulation. The second layer is a dense PyC coating; the outer layer is a corrosion resistant silicon carbon coating (SiC).

Boiling water is both the coolant and the main moderator in this reactor, although the carbon in the PyC and SiC provides some moderation as well. The fuel elements, containing 4.8% enriched uranium, are loaded at the top of the reactor core and are discharged at the bottom without the need for shutdown and depressurization.

As can be seen from Table 2, this reactor has very strong negative coolant temperature and void coefficients of reactivity. The fuel temperature reactivity coefficient is also strongly negative. Core reactivity is managed by means of movable gas-cooled control rods inserted from the core bottom. About 140–150 control rods with a spacing of about 12 cm are required for the reactor.

This core is designed as a frustum cone with the bottom being a perforated coolant dispenser, as shown in Figure 1 and the upper cap being a perforated plate that constrains the fuel particles. Therefore, the fuel is contained between the outer conical case and the perforated bottom and upper plates. The coolant flow path is as follows. Water coolant from jet pump nozzles enters the lower plenum, flows through the perforated coolant dispenser into the pebble bed. The water cools the pebble bed as it is heated and boils, while moving in the upward direction. The two-phase mixture exits the core through the perforations in the upper plate and enters the outlet plenum, located above the core. The cross section of the frustum cone increases vertically to compensate for void fraction increases and keeps the coolant velocity low. The balance of plant of the reactor is similar to standard BWR designs. The main core parameters for the reference 600-MWt reactor are reported in Table 2.

Table 2. Neutronic and Thermal-hydraulic Parameters of the Pebble Bed BWR

Fuel composition	UO_2
Fuel enrichment, weight percent ^{235}U	4.8
Dimensions	
Fuel diameter, mm	5.0
Buffer carbon layer, thickness, mm	0.5
Pyrolytic carbon layer, thickness, mm	0.5
Silicon-carbide layer, thickness, mm	0.5
Uranium loading in reactor core, kg	5,140
Fuel burnup, MWd/MT	40,000
Fuel residence time, year	1
Reactivity coefficients (@ 20,000 MWd/MT)	
Fuel temperature Doppler, 10E-5/°K	-3.9
Void, 10E-5/%void	-1690
Power defect, % Δk/k	7.6
Cold w/o xenon to Hot w/xenon, % Δk/k	-10
Specific power, kW/kg	117
Thermal /Electrical Power, MWt/ MWe	600/200
Core Dimension:	
Inlet Diameter, m	2.0
Outlet Diameter, m	2.49
Height, m	2.0
Coolant Parameters:	
Pressure, MPa	7.0
Hydraulic losses of the core, MPa	0.206
Inlet Temperature, °C	286
Number of particles in the core	7,122,000
Heat Flux, kW/m^2	132.6
Volumetric heat release, MW/m^3	95.5
Fuel temperature in center of the particles, °C	370
Thermal time constant, sec	0.86

The fuel temperature in the center of a 5 mm particle is only 85°C higher than the coolant temperature. The reactor response to operational transients, or to loss of coolant flow events is characterized by a strong negative temperature and void reactivity coefficients, as well as low thermal lag time for coolant temperature response to increases of the fuel temperature. These characteristics allow the pebble bed BWR to shut down rapidly and without active scram.

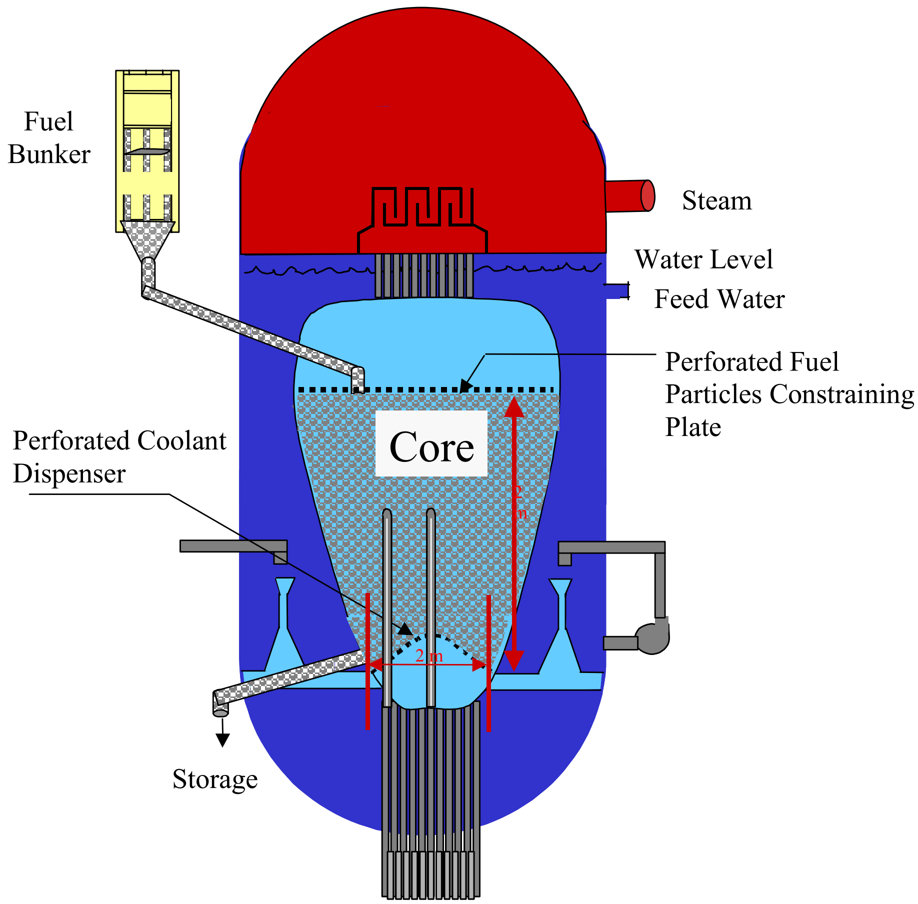


Figure 1. The Pebble Bed BWR.

The capability of TRISO fuel particle to retain the fission products at high temperature enhances the performance of the Pebble Bed BWR under severe accident conditions. Also, in case of complete loss of coolant the decay heat could be conducted radially across the core. It should be noted that the fission products silver and palladium diffuse through pyrolytic and silicon carbide coatings. In the gas reactors that have been operated to date, those fission products generally remained in the graphite matrix of the compacts. In this concept, they may be released to the coolant.

W2 – Pebble-Bed Reactor with Supercritical-Steam

This concept is virtually identical to the Pebble-Bed BWR except that the water coolant operates at supercritical pressures and temperatures. (Supercritical water-cooled reactors are also assessed in Appendix W5.) This eliminates the phase change within the core and the need for steam separators and dryers, as well re-circulation and jet pumps. Also higher thermal efficiencies (up to 45%) can be obtained with this approach. A schematic of this concept is illustrated in Figure 2.

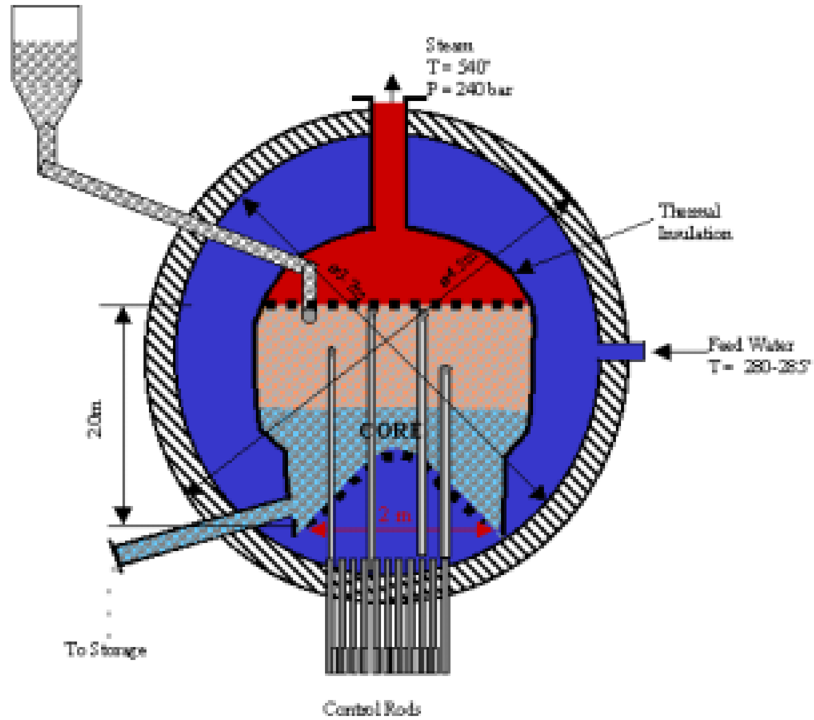


Figure 2. Pebble Bed Reactor with Supercritical Steam.

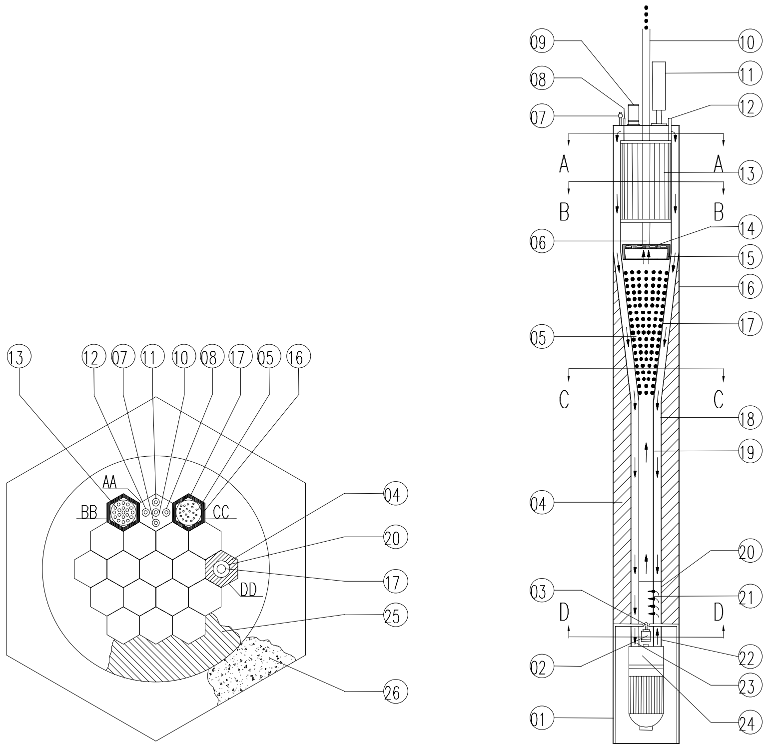
W7.2-b. Pebble Fuel Reactors with Zircaloy-Clad Fuel

W4 – Pebble Bed PWR

The reactor core is made of a variable number of modules, each generating about 1MWe. The basic module (see Figure 3) has the core region and a steam generator in its upper part, and a fuel chamber and pump in its lower part. The core consists of a 25-cm-diameter fluidizing tube in which, during reactor operation, the spherical fuel elements are kept in suspension by the upward coolant flow. The fuel chamber is a 10-cm diameter tube, which is directly connected underneath the fluidizing tube. A neutron absorber shell slides inside the fluidizing tube, acting similarly to a control rod, for the purposes of long-term reactivity control.

The operating pressure and temperature are the same as a traditional PWR. However, a steam generator of the shell-and-tube type is integrated into the upper part of each module. The pump circulates the water coolant inside the module moving upward through the fuel chamber, the fuel region, and the steam generator. Then the coolant flows back down to the pump through the concentric annular passage. Each module is provided with a pressurizer to keep the pressure constant.

The 8-mm diameter spherical fuel elements are made of slightly enriched uranium dioxide, clad with Zircaloy. The coolant velocity is selected to fluidize the particles so that the core operates at the reactivity maximum in the reactivity vs. moderator-to-fuel-ratio curve. That is, any deviation from the reference coolant flow level results in a reactivity decrease that automatically shuts down the reactor. In case of a complete loss of flow or coolant, the fuel particles fall down into the fuel chamber, which is a sub-critical configuration. Then the fuel chamber is cooled by natural convection transferring heat to the surrounding air or water pool.



(1) structural support, (2) hydraulic valve opener, (3) fuel discharge valve, (4) graphite jacket, (5) reactor core, (6) level limiter shaft, (7) depressurizer, (8) steam exit, (9) level limiter drive, (10) fuel feed, (11) pressurizer, (12) water entrance, (13) steam generator, (14) level limiter, (15) absorber shell, (16) hexagonal channel, (17) fluidization tube, (18) circular channel, (19) fuel chamber, (20) distributor, (21) entrance perforations, (22) coolant entrance, (23) coolant exit, (24) primary pump, (25) reflector, (26) biological shield.

Figure 3. The Pebble Bed PWR.

Online refueling of the modules is possible. The fresh fuel particles would be fed to the core region from the top of the module. The spent fuel leaves the module through a valve located at the bottom of the fuel chamber. The valve is operated by a hydraulic system allowing the spent fuel to be discharged from the fuel chamber into a permanently cooled storage tank.

It is proposed that the spherical pellets be produced from the existing cylindrical PWR pellets by an adequate grinding procedure.

W7.3. POTENTIAL FOR CONCEPT MEETING GENERATION IV GOALS

In the following sections, the Pebble Fuel Reactor (PFR) concept set is assessed against the Generation-IV goals. The advantages and/or disadvantages of the PFR concept set are evaluated relative to a typical Generation-III reactor (e.g., AP-600, ABWR, 80+), which serves as the reference system. In those areas for which no appreciable differences can be identified between the PFR concept set and the reference, the analyzed concept is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and metrics by means of a label in parenthesis.

3-a. Evaluation Against High Level Criteria

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

It is concluded that PFRs systems are equivalent to the reference ALWRs in the area of fuel utilization.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

The PFRs have the following advantage in the area of waste minimization and stewardship burden:

- Because of the higher surface-to-volume ratio, fuel pebbles operate at lower temperature in the repository, for given specific heat load. (SU2-3)

The PFRs with TRISO fuel have the following disadvantages in the area of waste minimization and stewardship burden:

- The high level waste volume will be larger than for the reference. (SU2-3)
- The long-term performance of SiC in a repository has not been established. (SU2-3)

It is concluded that PFR systems are substantially equivalent to the reference ALWRs in the area of waste minimization and stewardship burden although some uncertainties exist.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

PFRs exhibit the following disadvantages in the area of proliferation resistance:

- The presence of many fuel pebbles in the operations (thousands or even millions depending on the specific concept) in the reactor makes it more difficult to maintain fuel accountability. (SU3-1)
- It is possible to add special target pebbles for weapons material production. (SU3-1)
- It is not clear whether ceramic TRISO coated particles are easier or more difficult to recycle than the traditional UO_2 fuels with metallic cladding. However, it should be noted that relatively simple procedures and methods for recycling TRISO coated particles have been developed and used by General Atomics (recycle of product that did not meet specification during fuel manufacturing) and DOE. (SU3-3)

It is concluded that in terms of proliferation resistance, the PFR concepts may be somewhat worse than the reference ALWRs.

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

PFRs exhibit the following advantages in the area of safety and reliability under normal operating conditions:

- Because of the large surface-to-volume ratio the fuel operates at relatively low temperature. (SR1-2, SR1-3)
- The core can be refueled online, which eliminates the need for open-vessel refueling and might reduce the dose to workers. (SR1-1)

PFRs have the following disadvantages in the area of safety and reliability under normal operating conditions:

- Because these reactors operate with a fully fluidized bed and the silicon carbide coatings have a low fracture toughness, continuous collision of the fuel particles might lead to degradation of the fuel (e.g., cracking, erosion). (SR1-3)
- The silicon carbide coatings may be susceptible to irradiation-induced corrosion. (SR1-3)
- In the PFRs with TRISO fuel, cracking of the silicon carbide layers might lead to some modest water-carbon reactions. (SR1-2, SR1-3)
- Small fuel particles, or fragments of fuel particles, complicates the design of the grids that contain the fuel within the core. (SR1-3)
- The concept with Zircaloy-clad particles has a pump, a steam generator and a refueling machine for each fuel assembly, which significantly raises the number of components and the probability of a failure. (SR1-3)

The reliability of the particle fuel in a fully fluidized bed is the key unresolved issue associated with the PFRs. Therefore, the evaluators are concerned that these systems will perform worse than the reference ALWRs in terms of safety and reliability under normal operating conditions. Further research and development and demonstration are needed to show that this fuel is reliable.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

PFRs exhibit the following advantages in the area of safety and reliability under accident conditions:

- Upon loss-of-flow or loss-of-coolant events the fuel pebbles fall into a sub-critical configuration, thus yielding a passive scram. (SR2-3)
- Upon loss of the normal heat sink, removal of the decay heat from the core can be achieved through conduction radially across the core. (SR2-1)
- The TRISO coated fuel can retain nearly all fission products to relatively high temperatures (1600°C) that are above the temperatures expected during nearly all design basis accidents.

PFRs exhibit the following disadvantages in the area of safety and reliability under accident conditions:

- In the PFRs with TRISO fuel, cracking of the silicon carbide layers might lead to some water-carbon reactions. (SR2-2, SR2-3)

- The applicability of the existing correlations to the thermal-hydraulic phenomena occurring in the non-fixed core geometry during normal and accident situations (e.g., DNB, re-flooding) is not established. (SR2-2)

Because of the passive scram and decay-heat-removal features, it is concluded that the PFRs concepts will perform better than the reference ALWRs in terms of safety and reliability under accident conditions.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

PFRs exhibit the following advantage in the area of severe accident mitigation and need for offsite emergency response:

- The concepts with TRISO fuel particle have the capability of retaining nearly all the fission products at high temperature (i.e., 1600°C). (SR3-1)

It is concluded that the PFRs systems with TRISO coated fuel will perform better than the reference ALWRs in the area of severe accident mitigation and need for offsite emergency response. The Zircaloy clad pebble bed fuel should be equivalent to the reference ALWRs.

Economics–1. Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

PFRs exhibit the following advantages in the area of operating costs:

- Because the fuel pebbles randomly move within the core, the fuel is irradiated uniformly in the core without the need for shuffling. (EC3)

- Higher capacity factors can be achieved with online refueling. However, the committee believes that the overall reliability of the PFRs will be mainly determined by the integrity of the colliding fuel particles in the core. (EC3)

- For the concepts with TRISO fuel, the fuel cost might be somewhat smaller due to the elimination of pellet pressing, sintering, and grinding; manufacturing of zirconium-alloy tubes; end plug welding; grid manufacturing; and fuel assembly. (EC3)

PFRs have the following disadvantages in the area of operating costs:

- Large uncertainties exist on the possibility of consistently fabricating high-quality TRISO particles ten times larger than current gas reactor TRISO particles. (EC3)

The evaluators believe that at this point it is not possible to predict how PFRs will perform with respect to the reference ALWRs in terms of operating costs.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

PFRs have the following disadvantages in the area of capital cost:

- The concept with Zircaloy-clad particles has a pump, a steam generator and a refueling machine for each 1MWe fuel assembly, which will likely increase the overall capital costs. (EC1)
- Development costs will be large because these are revolutionary systems. (EC1, EC2)

On the other hand, the PFRs with TRISO fuel are basically equivalent to the reference ALWRs in terms of capital costs and financial risk, because they share similar designs for the primary and secondary systems.

3.b. Summary of the Strengths and Weaknesses

Strengths of the PFR concepts

- Passive scram upon LOCAs and LOFAs
- Potential for higher capacity factors from online refueling
- Potential for passive removal of the decay heat under accident conditions
- For the concepts with TRISO particle fuel, good retention of the fission products at high temperatures.

Weaknesses of the PFR concepts

- The performance of the fuel particles in a fully fluidized bed is unknown
- Accountability/diversion of fissile materials is an issue
- For the concept with one pump, one steam generator, and one refueling machine for each fuel assembly, the overall system reliability could be low and the specific capital costs could be high
- Fabrication of relatively large TRISO fuel particles might be difficult and costly.

W7.4. TECHNICAL UNCERTAINTIES

4.a. Research and Development Needs

The following research needs have been identified regarding the PFR concept:

1. It must be demonstrated that collisions between the fuel particles do not lead to erosion and/or cracking
2. It must also be demonstrated that irradiation-induced corrosion of the silicon carbide coatings will be predictable and acceptable

3. The coolability of the packed-bed geometry typical of the PFR concepts with TRISO fuel under accident conditions needs to be verified

4. Predictive tools for the relevant thermal-hydraulic and fuel behavior phenomena in the peculiar PFR geometry need to be developed and verified

5. Cost effective techniques to fabricate the fuel must be developed

6. For online refueling, effective and efficient means to measure the burnup of the fuel particles needs to be developed that can be used for discarding or re-injecting irradiated fuel in the core.

4.b. Institutional Issues - Licensability and Public Acceptance

The public could be receptive to the potential for passive scram and passive decay heat removal.

Licensability of these reactors (which are best characterized as revolutionary) depends mainly on the demonstration of adequate performance of the fuel particles under fully fluidized bed conditions.

4.c. Timeline for Deployment

The Pebble Bed Reactors can be developed within the Generation IV timeline; however, considerable fuel development and testing will be required.

W7.5. INITIAL ASSESSMENT: OBSERVATIONS AND CONCLUSIONS

The Technical Working Group believes that the potential of this concept set, especially the TRISO coated pebble bed BWR, for significant improvement in the safety area (deriving from the use of passive scram, passive decay heat removal, and fuel cladding materials that retain fission products at high temperatures) may not justify the anticipated large fuel development costs, and the uncertainties associated with the reliability of the particle fuel in water under fully-fluidized bed conditions.

W7.6. REFERENCES

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W7.7. Top-Tier Screening Sheet - Pebble Fuel Reactor Concept Set

Summary Evaluation: __X__ Retain ____ Reject

Goal		- -	-	+	++	Comments
SU1	Fuel Utilization					Thermal reactors with the same enrichment and burnup as reference LWRs
SU2	Nuclear Waste					-Concepts with TRISO fuel: unclear whether TRISO is a better waste form; high surface-to-volume ratio should prevent overheating in the repository -Concept with Zr cladding: same as reference LWRs
SU3	Proliferation Resistance					Accountability of the fuel particles. On-line refueling
S&R1	Worker Safety and Reliability					Cracking, erosion from particle collision. Irradiation induced corrosion
S&R2	CDF					Passive scram; passive decay heat removal
S&R3	Mitigation					-Concepts with TRISO fuel: good FP retention at high temperatures -Concept with Zr cladding: same as reference LWRs
E1	Life-Cycle Cost					Uncertainties in fuel-fabrication
E2	Capital Cost and Financial Risk					-Concepts with TRISO fuel: capital cost basically the same as reference LWRs; however, large development costs are expected. -Concept with Zr clad: large number of components per unit power

Appendix W8

Advanced Light Water Reactors with Thorium/Uranium Fuel Concept Set Report

December 2002

CONTENTS

FIGURES

TABLES

ABSTRACT

This appendix discusses the use of thorium in water-cooled, electric power producing reactors. Five general approaches are discussed: advanced light water reactors (ALWRs) with once-through seed and blanket thorium fuel, high conversion light water reactors with seed and blanket thorium fuel and U-233 recycle, ALWRs with once-through homogeneous thoria-urania (ThO_2-UO_2) fuel, ALWRs with once-through micro-heterogeneous thoria-urania fuel, and metal matrix thorium dispersion fuel.

The use of any of the proposed once-through thorium fuel cycles in a light water reactor will significantly improve the proliferation resistance of the fuel cycle and result in a much more durable high level waste form. Specifically, a once-through thorium fuel cycle will generate about 3 times less separable weapons material (plutonium) per kW-hr for the homogeneous approach and 4-6 times for the seed and blanket (heterogeneous) approach. The plutonium isotopics that are generated will be very "dirty," and the lifetime of the ThO_2--UO_2 fuels in a permanent repository will be much longer than for UO_2 (two orders of magnitude or slower dissolution rate after the canister and cladding has corroded away). The advantages of the high conversion light water reactors with seed and blanket thorium fuel and U-233 recycle are associated with their excellent long-term fuel resource sustainability. The energy potential of the U-233/Th-232 high conversion fuel cycle is expected to be on the order of 100 times that of the current all-uranium once-through fuel cycle. A near breeder design with a standard reactivity control and recycle of the blanket fuel is also expected to extend significantly existing natural uranium resources.

The key element of the ALWRs with once-through seed and blanket thorium fuel is the seed and blanket unit developed by Alvin Radkowsky and coworkers with a well-moderated seed region and a slightly under-moderated blanket region. This arrangement provides the necessary flexibility for designing the seed as an efficient supplier of neutrons to a sub-critical blanket that, in turn, is designed for an efficient generation and burning of U-233. The high conversion light water reactors with seed and blanket thorium fuel and U-233 recycle are similar to current pressurized water reactors, but with a core design that conserves neutrons and breeds U-233. Specifically, separate seed and blanket fuel regions are used to maximize the neutron production, the reactor is controlled by moving the seed rather than inserting absorber rods so as to eliminate parasitic neutron losses, blankets and reflectors are located to minimize leakage, and the fuel rods are spaced relatively closely.

The third approach discussed in this appendix for using thorium in LWRs is the use of high burnup homogenously mixed thorium-uranium dioxide (ThO_2-UO_2) fuels in ALWRs. In this case the thoria and urania are mixed uniformly, and the fuel rods and bundles have essentially the same geometry as current LWR fuel. A variation on this approach is some small amount of "micro-heterogeneity." Here the fuel form might be a fuel rod with alternating short stacks of thoria and urania pellets, or it might be alternating thoria and urania fuel rods. Providing some small separation between the uranium and thorium improves the core reactivity and burnup. The fifth approach is to use a fuel composed of a fine dispersion of thoria and urania in a metal matrix.

In assessing the strengths and weaknesses of these concepts, it is concluded that the cost penalties associated with the use of the various homogeneous thorium fuel cycles discussed in this appendix may prevent their introduction in the near future. However, farther out in the future our high-grade uranium ore supplies will become depleted and yellow cake prices will rise and the thorium/uranium fuel cycles with improved uranium utilization will eventually become cost effective.

The significant advantages of the once-through thorium cycles with respect to proliferation resistance and waste form stability are very attractive to the Federal government and society as a whole, but provide little incentive to the current nuclear fuel industry. The energy resource sufficiency advantage of the U-233/Th-232 high conversion reactor fuel cycle is currently outweighed by proliferation, reliability, and cost issues. The metal matrix fuels are relatively undeveloped.

W8.1 INTRODUCTION

This appendix discusses the use of thorium in advanced light water reactors (ALWRs). Five general approaches are discussed: ALWRs with once-through seed and blanket (Radkowsky) thorium fuel, high conversion light water reactors with seed and blanket thorium fuel and U-233 recycle, ALWRs with once-through homogeneous thoria-urania (ThO_2-UO_2) fuel, ALWRs with once-through micro-heterogeneous thoria-urania fuel, and metal matrix thoria-urania fuel. The potential benefits of the Radkowsky seed and blanket, the homogeneous and micro-heterogeneous thoria-urania, and the metal matrix *once-through* fuel cycles are discussed in the following paragraphs. Background information about the Radkowsky seed and blanket, the homogeneous and micro-heterogeneous thoria-urania, and the metal matrix once-through fuel cycles is presented in Sections W8.1a, c, and d below. The advantages of the U-233/Th-232 light water breeder reactor thorium fuel cycle are discussed in Section W8.1b of the introduction.

Proliferation Resistance. LWRs generate plutonium from U-238 neutron capture. Today, worldwide, there are about 300 tons of separated civilian plutonium, primarily in France, Germany, the United Kingdom, Russia, and Japan. In addition, more than 1000 tons of plutonium is contained in spent LWR fuel worldwide. It takes only about 5 to 6 kilograms of Pu-239 to build a weapon (Mark 1992). Currently, the fissile materials within the civilian power programs are adequately safeguarded by an effective international system. However, it would be beneficial if future nuclear fuel cycles and nuclear materials safeguards systems, in combination, could provide an even higher degree of resistance to nuclear material proliferation or diversion.

High burnup thorium fuels will improve the weapons material proliferation-resistance of LWRs in two ways. First, there will be 3 to 6 times less separable weapons material (plutonium) generated per kW-hr because most of the fertile material will be thorium. Second, the isotopic content of the remaining plutonium will be much less desirable for use in weapons. LWR fuel that is taken to high burnups contains plutonium isotopes (primarily the even numbered isotopes such as Pu-238) that make it much more resistant than lower burnup fuel to nuclear weapons proliferation. Pu-238 is primarily produced in a three-step neutron absorption in U-235, and its fraction increases approximately with the square of the fuel burnup. Thus, more Pu-238 is generated both due to a higher initial content of U-235 in the UO_2 driver (~20% in comparison with 5% for typical UO_2 fuels) and due to the higher burnup. The even numbered plutonium isotopes in LWR spent fuel release spontaneous neutrons that significantly decrease the probable yield of a nuclear weapon. They also release significant heat that makes design and fabrication of the weapon difficult.

High-Level Waste Form. Because ThO_2 is the highest oxide of thorium, while UO_2 can be oxidized further to U_4O_9, U_3O_8, and UO_3, ThO_2-UO_2 fuel appears to be a much better waste form than conventional UO_2 fuel. The lifetime of ThO_2-UO_2 fuel (exposed to wet air oxidation) appears to be significantly longer than for UO_2 (two orders of magnitude or greater slower dissolution rate after the canister and cladding has corroded away).

Improved Nuclear Power Plant Economics. The burnup-related reactivity swing in a ThO_2-UO_2-fueled reactor is smaller than in a UO_2 core because of the high conversion ratio of the thorium. Most of the U.S. plants are currently operating with 18-month or longer fuel cycles (mostly limited by the USNRC burnup limits). With improved burnup capacity fuel, many of these plants could reduce the number of fuel assemblies loaded in each cycle, thus reducing fuel costs and making nuclear energy more competitive.

W8.1.a. Advanced Light Water Reactors with Once-Through Seed and Blanket Thorium Fuel

There are a number of ways thorium can be used in LWRs. Probably the best-known once-through thorium fuel-cycle concept was developed by Dr. Alvin Radkowsky and associates in Israel and is known as the Radkowsky Thorium Fuel Cycle (Galperin et al. 1999). The concept is based in part on the ideas and experiences of the Bettis Atomic Power Laboratory's Light Water Breeder Reactor (LWBR) program as implemented and successfully demonstrated at the Shippingport reactor in the 1980s. However, in contrast to the LWBR, the Radkowsky concept assumes a once-through thorium fuel cycle with no recycling; the U-233 that is bred is mostly burnt *in situ*, and the fuel rods that contain the U-233 (which is denatured by nonfissile uranium isotopes) are then disposed of.

The main idea of the Radkowsky thorium fuel cycle is the utilization of a seed-blanket unit (SBU) that is fully interchangeable with current LWR fuel bundles. The SBU geometry allows a spatial separation of the uranium (mostly in the seed) and thorium (blanket) parts of the fuel bundle. The central region of the assembly (seed) includes uranium enriched to a maximum of 20%, while the external region of the assembly (blanket) includes natural thoria (ThO_2) spiked by a small amount of 20% enriched urania (UO_2). This arrangement provides the necessary flexibility for designing the seed as an efficient supplier of well-thermalized neutrons to a sub-critical blanket that, in turn, is designed for efficient generation and in-situ burning of U-233. This approach has been applied to both Russian designed water-cooled, water-moderated energy reactor (VVER) and pressurized water reactor (PWR) core designs with considerable success and could also be applied to other water reactor designs in the future (e.g., boiling water reactors (BWRs) or small modular light water reactors currently under development). One variant of this approach uses plutonium rather than uranium as fuel (Galperin et al. 2001). This improves the nonproliferation characteristics of the concept by virtue of being able to dispose of large amounts of plutonium.

The Radkowsky thorium fuel project began in 1994 with initial studies funded by the Radkowsky Thorium Power Corp. (RTPC). In 1996 the program received funding from the DOE-NN Industrial Partnering Program (IPP) (now the Initiatives for Proliferation Prevention Program), and expanded significantly with the inclusion of the Brookhaven National Laboratory (BNL), the Massachusetts Institute of Technology (MIT), and a team of Russian participants led by the Russian Research Center-Kurchatov Institute (RRC-KI). DOE/IPP has awarded two subsequent grants to this group of organizations. The objective of the current project is to develop and demonstrate key elements of the Radkowsky thorium fuel cycle concept for implementation in commercial PWRs and VVERs.

The work by the Western organizations has been supplemented by work on a variant of the PWR design funded by a FY-00 DOE-NE NERI grant. In that project, Optimization of Heterogeneous Schemes for the Utilization of Thorium in PWRs to Enhance Proliferation Resistance and Reduce Waste, the objective is to look at a core where the seed and blanket regions are each the size of an assembly with the loading of the assemblies throughout the core in either a checkerboard or "dispersed" pattern (Wang et. al. 2000; Todosow et al. 2001). This approach is to be compared with that described above where each assembly is an SBU. In addition to BNL and MIT, the NERI project has recently added collaborators from Kyung Hee University, Korea Advanced Institute for Science and Technology (KAIST) and Korea Advanced Energy Research Institute (KAERI).

W8.1.b. High Conversion Light Water Reactors with Seed and Blanket Thorium Fuel and U-233 Recycle

LWRs attained economic significance during the mid-1960s for central power station electricity generation on the basis of relatively low capital and uranium costs, abundant enrichment capacity, and strong technical support from the U. S. Naval Reactor Program. However, the subsequent development sequence of nuclear power in the world was not what had been originally envisioned. Originally, it was expected that a modest number of LWR plants would be built, providing needed power, the technical basis for a growing nuclear industry, and the fuel for fast spectrum breeder reactors. The fast spectrum breeder reactor was expected to provide the basis for a fuel self-sufficient (plutonium recycle based) nuclear power industry (Shapiro et al. 1977). However, the commercial breeder reactor was not fully developed and the LWR was a much stronger commercial competitor for power plant construction versus fossil fuels (in the 1970s and early 1980s) than originally expected. The result is that, worldwide, we have a large number of LWRs without a long-term sustainable fuel cycle. The current once-through uranium fuel cycle is essentially transitory, i.e., it had a beginning and it will end not too far in the future.

However, it is possible to design and build a thermal spectrum LWR with a fully self-sufficient fuel cycle if the U-233/Th-232 fuel cycle is adopted (Hecker and Freeman 1981). The primary advantage of the use of U-233 fissile material in thermal reactors is that the average number of neutrons produced per atom of fissile material destroyed is large enough for fuel self sufficiency, whereas, if either U-235 or Pu-239 is used in a thermal spectrum reactor the average number of neutrons produced per atom of fissile material destroyed is too small for breeding. The Th-232 is needed to produce the U-233, of course. The high conversion light water reactors with seed and blanket thorium fuel and U-233 recycle can take advantage of all the technology that has been developed to support the PWRs. However, their core designs must be slightly different so as to better conserve neutrons (Conners et al. 1979). Specifically, separate seed and blanket fuel regions are used to optimize the neutron economy, the reactor is controlled by moving the seed (with PWR type control rod drives) rather than inserting absorber rods so as to eliminate parasitic neutron losses, blankets and reflectors are located to minimize leakage, and the fuel rods are spaced relatively closely.

Thorium, which averages 7.2 parts per million in the earth’s crust, is the 39th most abundant of the 78 crustal elements. It is about three times more abundant than uranium. When bred to the fissile U-233, thorium releases about the same energy per unit mass (79 TJ_{th}/kg) as uranium bred to Pu-239 (80.4 TJ_{th}/kg). Thorium and its compounds have been produced primarily from monazite, where it is produced as a by-product of the recovery of titanium, zirconium, tin, and rare earths. Only a small portion of the thorium produced has been consumed. Limited demand for thorium, relative to the demand for rare earths, has continued to create a worldwide oversupply of thorium compounds and mining residues. Thus, in the short term, thorium is available for the cost of extraction from rare-earth processing wastes. In the longer term, large quantities of thorium are available in known monazite deposits in India, Brazil, China, Malaysia, and Sri Lanka.

The existing LWRs convert some fertile U-238 or Th-232 into fissile fuel, however, the overall nuclear resource utilization is only about 1 percent of the energy potentially available from the mined ore. A comparison of the energy potentially obtainable from the current world-wide thorium resources and use of LWBRs with the energy available from the once through LWR fuel cycle in the existing LWRs, and from known fossil reserves is shown in Figure 1 to the right (Hecker and Freeman 1981). Based on the use of a well-established and successful LWR technology and the potential for an assured energy supply for a very long time period, the development of the LWBR U-233/Th-232 fuel cycle appears to be an attainable and important alternative for future energy generation.

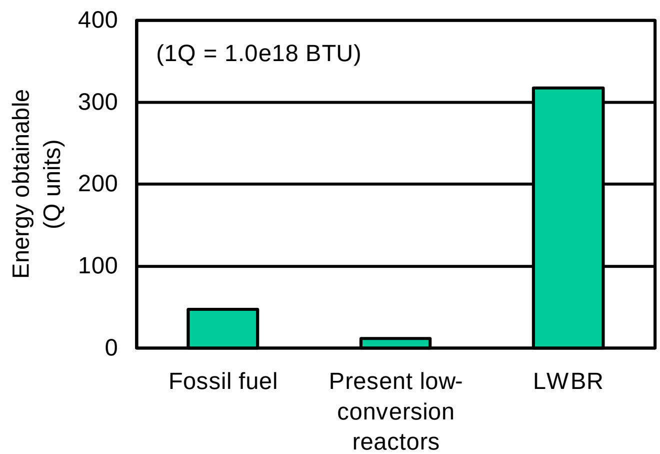


Figure 1. Comparison of the energy potential in fossil fuel, the current once-through LWR fuel cycle, and the LWBR (Hecker and Freeman 1981).

W8.1.c. Advanced Light Water Reactors with Once-Through Homogenous and Micro-Heterogeneous Thoria-Urania Fuel

A third approach for using thorium in current LWRs or ALWRs is the use of high burnup homogenously mixed thorium-uranium dioxide (ThO_2-UO_2) fuels. In this case the thoria and urania are mixed uniformly, and the fuel rods and bundles have essentially the same geometry as current LWR fuel (Herring et al. 2001). Fuel with 75% thoria and 25% urania (enriched with U 235 to slightly less than 20%) can reach burnups of about 54 MWd/kg initial-heavy-metal. Fuel with 65% thoria and 35% urania can reach burnups of about 75 MWd/kg. A variation on this approach was first developed during the LWBR program and more recently investigated at MIT and includes some small amount of what is called “micro-heterogeneity.” Here the fuel form might be a duplex pellet with the urania on the inside and the thoria on the outside, or it might be a fuel rod with alternating short stacks of thoria and urania pellets, or it might be alternating thoria and urania fuel rods (Zhao et. al. 2001). Providing some small separation between the uranium and thorium improves the core reactivity and achievable burnup.

These approaches are being investigated in a FY-99 DOE-NE NERI project entitled “Advanced Proliferation Resistant, Lower Cost, Uranium-Thorium Dioxide Fuels for Light Water Reactors” (MacDonald 2000, 2001a, 2001b, and 2001c). The NERI project is funding work at two DOE national laboratories (Idaho National Engineering and Environmental Laboratory and Argonne National Laboratory), three universities (MIT, Purdue University, and University of Florida), and two fuel vendors (Framatome Technologies and Westinghouse). Siemens is involved in the project as a reviewer and KAERI is also a participant in the project. The project has been organized into four tasks:

- A neutronics and economics analysis to determine the economic viability of various ThO_2/UO_2 fuel designs in PWRs
- An assessment of whether or not ThO_2/UO_2 fuel can be manufactured economically
- An evaluation of the behavior of ThO_2/UO_2 fuel during normal, off-normal, and accident conditions and a comparison of the results with the results of previous UO_2 fuel evaluations and U.S. Nuclear Regulatory Commission (NRC) licensing standards
- An assessment of the long-term stability of ThO_2/UO_2 high-level waste.

The results of this work will be discussed in Sections W8.2c and W8.2d below.

W8.1.c. Metal-Matrix Thoria-Urania Dispersion Fuel

This fuel is composed of a fine dispersion of thoria-urania micro-spheres in a zirconium metal matrix. Because of the improved stability of this fuel during irradiation, it is suitable for very high burnup use. The spent fuel is highly proliferation resistant and a relatively good waste form.

W8.2 CONCEPT DESCRIPTIONS

The ALWRs with once-through seed and blanket thorium fuel are discussed in Section W8.2-a below, the high conversion light water reactors with seed and blanket thorium fuel and U-233 recycle are discussed in Section W8.2-b, ALWRs with once-through homogeneous thoria-urania fuel are discussed in Section W8.2-c, ALWRs with once-through micro-heterogeneous thorium fuel are discussed in Section W8.2-d, and the metal-matrix thoria-urania dispersion fuel is discussed in Section W8.2-e. The developers of the concepts primarily wrote the concept summaries reported below. They have been edited for style and brevity. Some of their statements may not reflect the judgment of the Technical Working Group, which is reported instead in Section 3 of this appendix.

W8.2.a. Advanced Light Water Reactors with Once-Through Seed and Blanket Thorium Fuel

As stated above, a key element of the concept is the SBU fuel assembly geometry. The SBU geometry allows a spatial separation of the uranium (mostly in the seed) and thorium (blanket) parts of the fuel. The central region of the assembly (seed) includes uranium metal or uranium oxide fuel enriched to a maximum of 20% U-235, while the external region of the assembly (blanket) includes natural thorium dioxide spiked by a small amount of 20% enriched UO_2. This arrangement provides the necessary flexibility for designing the seed as an efficient supplier of well-thermalized neutrons to a sub-critical blanket that, in turn, is designed for an efficient generation and in situ burning of U-233. The initial uranium content of the blanket provides power production in that region until sufficient U-233 has been produced, and also denatures the bred U-233.

The spatial separation of the seed and blanket sub-assemblies results in a different lattice design: a well-moderated seed region ($Vm / Vf = 3.3$) and an under-moderated blanket region ($Vm / Vf = 1.8$). Figure 2 is a diagram showing the fuel assembly layout that would replace a 17x17 assembly in a Westinghouse type reactor. (Studies to date have focused on this assembly design, however, the latest design does not have separating wall between seed and blanket and it makes the thermal-hydraulic performance worse.) The guide tubes shown are in identical locations to that of a normal PWR assembly.

Within the seed, the power density is high leading to the use of an annular metal (U/Zr alloy) or oxide fuel rod clad in Zircaloy (Busse and Kazimi 2000, Wang et al. 2001). The high thermal conductivity (in the case of the metal fuel) and annular geometry of the fuel keeps the average fuel temperature down within acceptable limits. The thermal-hydraulic analysis done to date has also shown an acceptable departure-from-nucleate-boiling ratio (>1.3) and maximum fuel temperature. The meeting of these limits is particularly important because of the relatively high power density in the seed. The blanket fuel is oxide and also clad in Zircaloy and the thermal hydraulic limits are more easily achieved.

One of the novel features of the Radkowsky thorium fuel cycle is its in-core fuel management scheme. The standard multi-batch fuel management of a PWR is replaced by a scheme that is based on two separate (seed and blanket) fuel flow routes. Basically, seeds are treated similarly to the standard

PWR assemblies, i.e., approximately one-third of the seeds are replaced periodically by "fresh" seeds, and the remaining, partially depleted seeds are reshuffled together with partially depleted blankets to form a reload configuration for the next cycle.

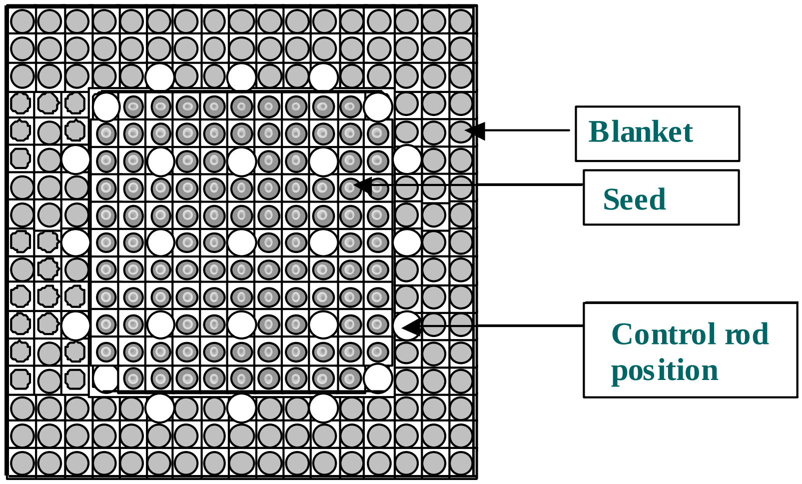


Figure 2. Radkowsky seed and blanket unit for PWRs.

For reasons of fuel economy the thorium blanket in-core residence time is quite long (~9 years). A long residence time is required to achieve a large accumulated burnup for the thorium part of the fuel, about 80 MWd/kg, or ~9 MWd/kg for each cycle, assuming the blanket is removed after nine annual or six 18-month cycles. Significant irradiation testing of the thoria-urania fuel would be required to confirm this high burnup capability.

The variant of the Radkowsky thorium fuel cycle known as the Radkowsky Thorium Fuel Plutonium Incinerator (RTPI) replaces uranium in the seed with Pu. The current generation of PWRs designed to use mixed uranium-plutonium oxide (MOX) fuel to eliminate weapons-grade plutonium have three noticeable drawbacks, namely, a relatively low annual plutonium elimination rate, a reduced control rod worth, and deterioration of the safety characteristics, in particular, the moderator temperature coefficient for higher plutonium content. The later two limit the amount of MOX that can be loaded into a PWR. The proposed RTPI alleviates these shortcomings. The annual elimination of plutonium is three times that of a PWR using MOX fuel. Furthermore, there is no degradation of control rod worth in the RTPI and the moderator temperature coefficient is more negative in thorium fuels. The increased rate of plutonium burning may be attributed to the fact that only 9% of the RTPI fertile component is U-238. The modest impact on control characteristics is due to the heterogeneous (SBU) nature of the assembly, which allows for a high moderator-to-fuel ratio in the seed, which, in turn, restores the reactivity worth of the control rods that would otherwise be reduced due to the presence of Pu. The more negative moderator temperature coefficient comes from the reduced resonance integral and higher fast fission threshold in the Th-232 compared to the U-238. In addition, the RTPI residual plutonium (i.e., the plutonium discharged from the core) is less usable in a weapon than the MOX residual.

Another variant of the Radkowsky Thorium Fuel Cycle is the whole assembly seed and blanket option. For this design, the seed and blanket are *each* the size of a PWR assembly and are distributed in the core in a checkerboard pattern. The seed fuel is UO_2. Again the blanket is mixed thoria-urania with only a small amount of U-238 present to denature the U-233 produced. In order to flatten the power sharing between seed assemblies and blanket assemblies, burnable poison (Er_2O_3) is added into the central void of the seed fuel pellets. The whole assembly seed and blanket concept is designed for an 18-month fuel cycle.

The SBU design for the VVER reactors has two major changes from the PWR design: a triangular pitch is used (which is the standard pitch in a VVER) and the driver or seed fuel rods are a 3-petal twisted rod self-spacing design (technology based on the Russian submarine program). The VVER seed rods are shown in Figure 3. Figure 4 is a sketch of the VVER SBU assembly arrangement.

Figure 3. VVER seed fuel rods.

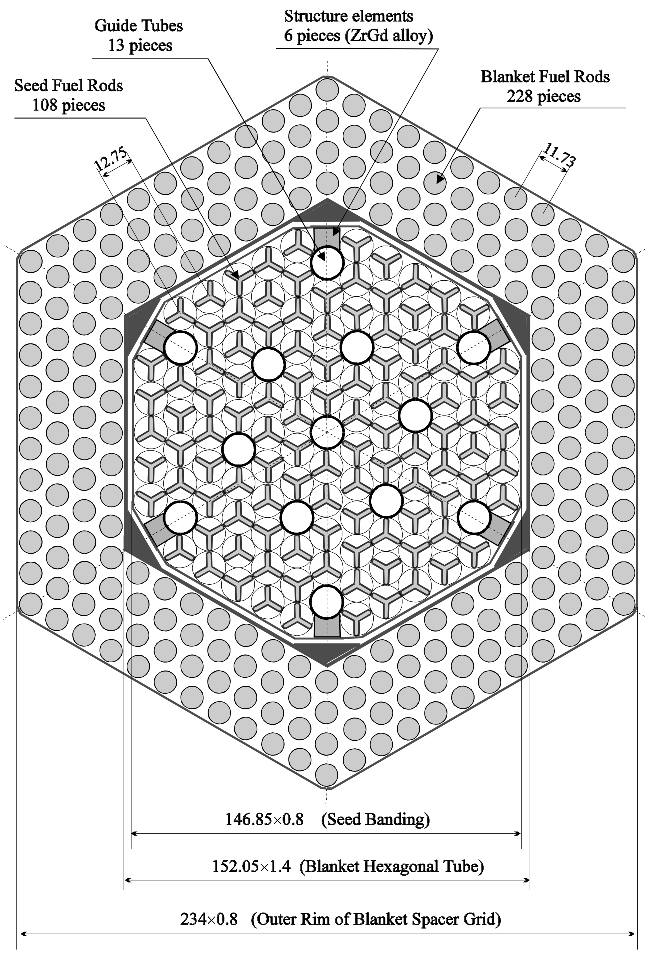


Figure 4. Seed blanket unit for VVERs.

W8.2.b. High Conversion Light Water Reactors with Seed and Blanket Thorium Fuel and U-233 Recycle

A small, light water-cooled breeder reactor with U-233/Th-232 fuel was developed and demonstrated by the U. S. Naval Reactors Program. The LWBR was operated at the Shippingport Atomic Power Station, which was a Department of Energy (DOE) (formerly Atomic Energy Commission)-owned nuclear plant. The LWBR core was developed for operation within the constraints of the relatively small Shippingport plant. However, the interior modules were designed so that they could be used directly in a large high conversion reactor core (Conners et al. 1979; Cambell et al. 1987; Hecker 1979; Hecker and Freeman 1981; Sarber et al. 1976).

The nuclear design of the LWBR core utilized a seed-blanket concept similar to that successfully applied to the first two PWR cores operated at Shippingport, but with the reactivity control provided by core geometry changes (movable seed fuel) instead of poison rods. Figure 5 shows the arrangement of the core components in the Shippingport reactor vessel. Figure 6 shows a plan cross-section of the LWBR core.

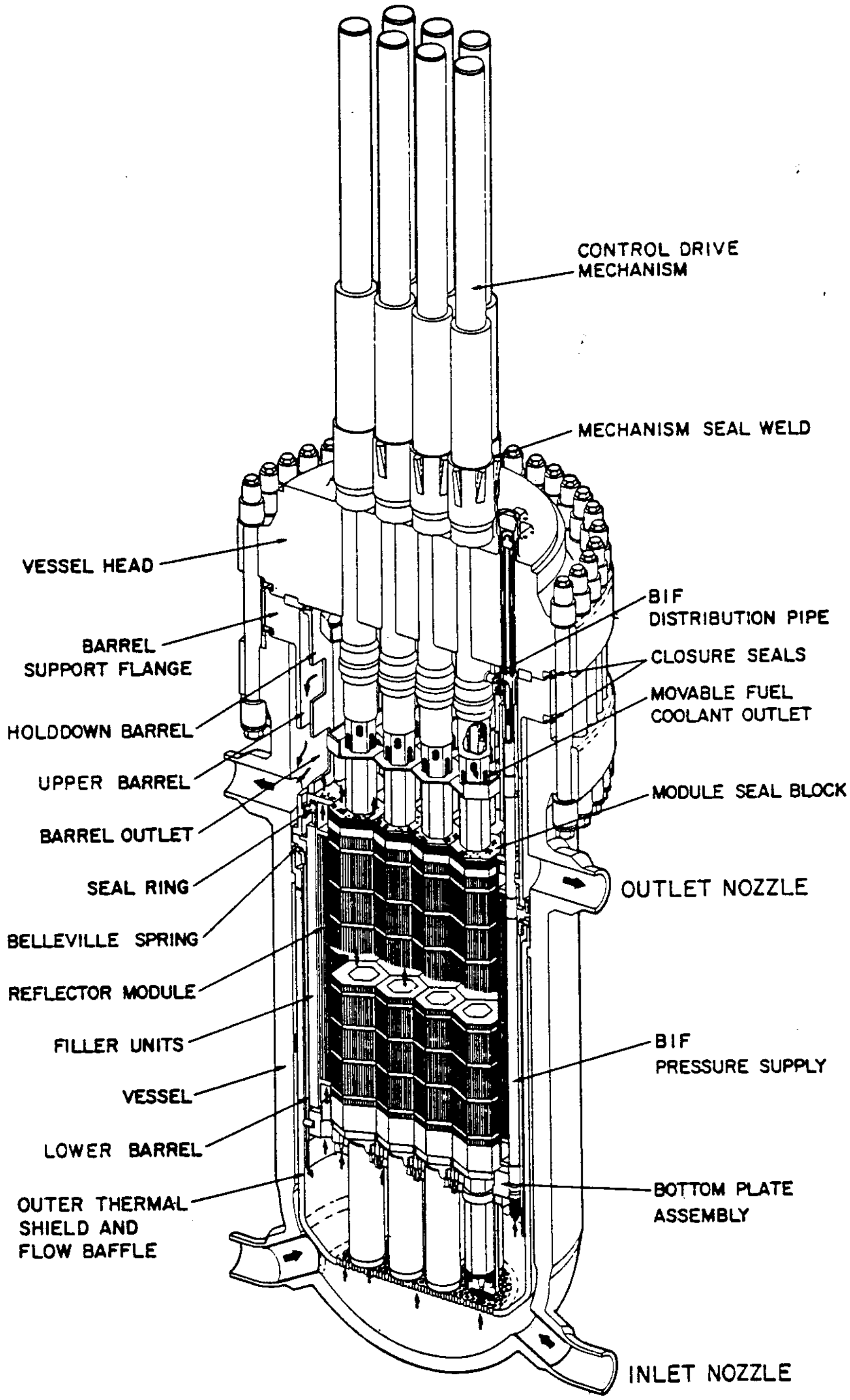


Figure 5. LWBR core in the Shippingport reactor vessel (Conners et al. 1979).

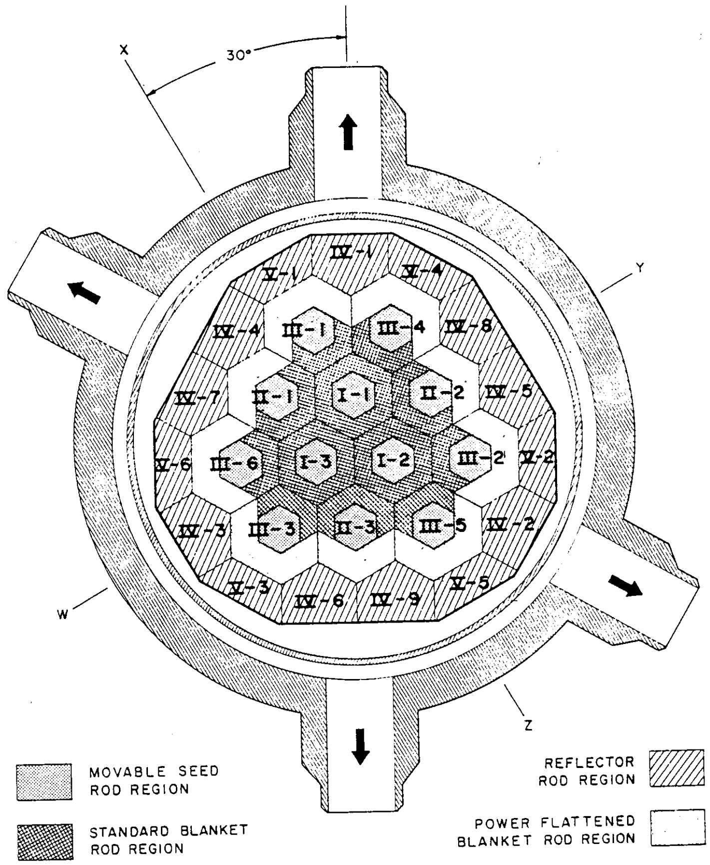


Figure 6. Cross-section of the LWBR core in the Shippingport plant (Sarber 1976).

The LWBR core was designed to minimize parasitic neutron absorption in core and structural materials. The core design features that contributed to the improved neutron economy in the Shippingport LWBR included:

- Seed and blanket regions tailored to maximize neutron production
- Movable seed fuel to control core reactivity, rather than conventional poison control rods, soluble poison, or burnable poison
- Peripheral radial and axial thoria blanket regions to reduce neutron leakage from the core.
- Reflectors to reduce neutron leakage from the core
- Zirconium alloy material for the fuel rod cladding and for most of the structures in the active fuel region
- A relatively tight fuel pitch.

The four primary fuel regions (seed, standard blanket, power-flattening blanket, and reflector blanket) were each optimized to maximize neutron absorption in the thorium and to minimize neutron losses.

The LWBR core in Shippingport was fueled with Th-232 and U-233 oxide fuel rods clad with Zircaloy-4. The seed fuel rods had an outside diameter of 0.306 inches and a triangular pitch of 0.369 inches. The standard blanket rods had an outside diameter of 0.572 and a pitch of 0.6304 inches (a considerably lower fuel to water ratio). The power flattening blanket rods were similar to the standard blanket rods and the reflector blanket rods had a relatively large diameter of 0.832 and a pitch of 0.901 inches. The U-233 enrichment in the UO_2 was about 98% and the fraction of UO_2 in the ThO_2/UO_2 seed rods varied over four radial regions from 4.4 to 5.3% (effective heavy metal enrichments of 4.3 to 5.2%). The U-233 enrichment in the standard blanket rods varied over five radial regions from 1.2 to 2.0% of the total heavy metal. In addition, the ThO_2/UO_2 stack lengths and fissile loadings varied axially in both the seed and blanket rods (there was ten inches of pure ThO_2 reflector on each end of all the seed and standard blanket rods plus three radial regions with additional thoria steps in both the seed and standard blanket assemblies). The reflector rods were made of pure ThO_2. It is likely that a modern core design for a large LWBR could be much less complex than the Shippingport design, never the less, some radial and axial zoning would probably be needed to conserve neutrons.

Recycling of thoria-urania fuels has been demonstrated using the Thorex process. However, the relatively poor extraction properties of thorium nitrate require considerably higher acid concentrations than the Purex process for uranium fuels and a throughput reduced by about half in a given size plant (Wilson 2000). The high acid concentrations raise corrosion issues that need to be addressed. However, the principal drawback to the recycle of U-233 is the presence of hard gamma emitters (0.7 to 2.6 MeV) among the descendents of the U-232 that is formed from (n, 2n) reactions with both the Th-232 and U-233 (Shapiro et al. 1977). The U-232 is an alpha emitter with a 72-year half-life, which is always present at concentrations of tenths to hundreds of a ppm after one cycle and can reach concentrations of 7,000 to 11,000 ppm after four or five passes through the core (Shapiro et al. 1977). If Th-230 is present in the ore, then even higher U-232 contamination levels will be reached. Therefore, U-233 enriched fuels must be manufactured remotely in gamma-shielded environments (hot cells), a relatively expensive operation. The high neutron and gamma activity of thorium recycle fuels will also complicate procedures outside the fabrication facility. The fresh U-233 enriched recycle fuels will need more shielding than mixed uranium-plutonium oxide fuels, and fuel handling and inspection before core loading will be more difficult. Special fresh fuel storage and handling areas may need to be constructed (Shapiro et al. 1977).

One other issue that is unique to the use of U-233/Th-232 fuel is the shutdown reactivity addition due to the decay of Pa-233 and the buildup of U-233 after shutdown (Shapiro et al. 1977).

One final point, the high level waste form the U-233/Th-232 fuel cycle will contain fewer long lived minor actinides than plutonium recycle waste (Lung 1996).

W8.2.c. Advanced Light Water Reactors with Once-Through Homogeneous Thoria-Urania Fuel

As mentioned in the introduction, in this approach the thoria and urania are mixed uniformly, and the fuel rods and bundles have essentially the same geometry as current LWR fuel. Fuel with 75% thoria and 25% urania (enriched with U-235 to slightly less than 20%) can reach burnups of about 54 MWd/kg initial-heavy-metal. Fuel with 65% thoria and 35% urania can reach burnups of about 75 MWd/kg. Figure 7 shows the reactivity versus burnup of a 25%UO_2-75%ThO_2 core and an all UO_2 core (both cases with no burnable poison). Notice that the reactivity swing of the thoria-urania core is significantly less than that of an all-uranium core, but the reactivity of the thoria-urania core is also less than that of the all-uranium core. This suggests that the enrichment and natural uranium requirements for a thoria-urania core will be somewhat higher than for an all-uranium core, but the burnable poison needed will be far less. In fact, it may be possible to avoid the use of soluble boron in a thoria-urania core.

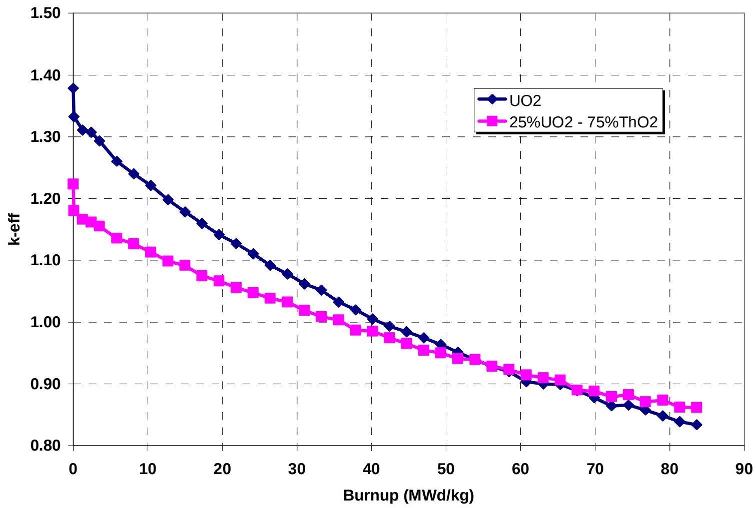


Figure 7. Multiplication Factor versus burnup of a 25%UO_2-75%ThO_2 pin cell, and an all UO_2 pin cell (MacDonald et al. 2000).

W8.2.d. Advanced Light Water Reactors with Once-Through Micro-Heterogeneous Thoria-Urania Fuel

A variation on the homogeneous approach was developed at MIT [based on work in the LWBR program and previous studies (Radkowsky, 1990)] and includes some small amount of what is called "micro-heterogeneity" (Zhao 2001). In this case the fuel form might be a duplex pellet with the urania on the inside and the thoria on the outside, or it might be a fuel rod with alternating short stacks of thoria and urania pellets, or it might be alternating thoria and urania fuel rods. These concepts are illustrated in Figure 8 below.

Three primary variants of micro-heterogeneity have been investigated to date: (1) duplex fuel where each pellet is composed of a center of UO_2 surrounded by a ThO_2 annulus or vice versa , (2) axial micro heterogeneity where pellets of UO_2 are sandwiched between ThO_2 or ThO_2-UO_2 pellets in a typical PWR fuel pin geometry, and (3) various arrays of single UO_2 or ThO_2 pins. Typical reactivity limited batch burnup results are presented in Table 1 below (MacDonald et al. 2001c). In each case shown below, the thoria-urania fuel contained 35%UO_2 and 65%ThO_2 and burnup stopped at a k-infinity of 1.03.

As shown in Table 1 below, the homogeneous thoria-urania fuel is only able to reach about 90% of the burnup of the reference UO_2 core. The duplex fuel pellet with the ThO_2 on the outside provides about an 11% improvement in burnup over the homogeneous thoria-urania fuel option, but about the same burnup as the UO_2 fuel currently used in LWRs. Only the axially micro-heterogeneous ThO_2/UO_2 fuel, with pure ThO_2 in the blanket region (no denaturing of the thoria), increases the fuel discharge burnup a significant amount over the UO_2 base case, about 13 to 15% for the cases analyzed. These effects are achieved due to a combination of changes in cross-shielding, conversion ratio, and local fissile worth, where local fissile worth is mainly responsible for the "burnable poison effect" at beginning-of-cycle, the conversion ratio causes burnup-related effects, and cross-shielding is responsible for the spatial effects.

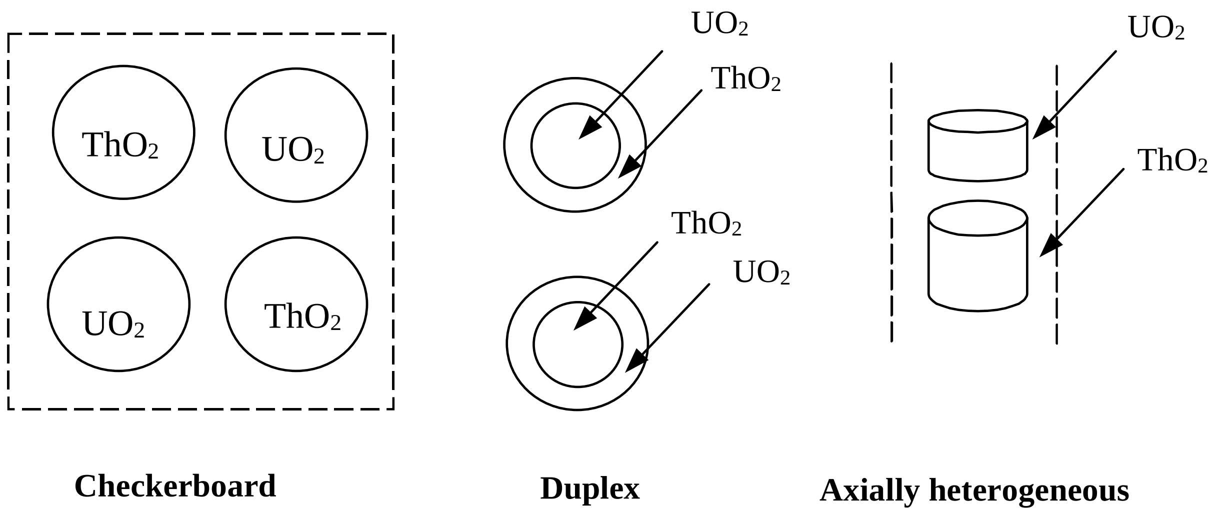


Figure 8. Representative configurations of micro-heterogeneous ThO_2/UO_2 fuel.

Table 1. Batch burnup available from various micro-heterogeneous fuel types.

f Fuel Type	Batch Burnup	Percent increase over Homogeneous ThO_2/UO_2 Fuel	Percent increase over All-UO_2 Base Case
All-UO_2 Reference Base Case	53.55	11	
Homogeneous ThO_2/UO_2 fuel	48.16		-10
Duplex, ThO_2 inside	48.49	1	-10
Duplex, ThO_2 outside	53.57	11	-
Axial micro-heterogeneous, 2 cm of ThO_2 and 1 cm of UO_2	57.10	19	7
Axial micro-heterogeneous, 8.2 cm of ThO_2 and 4 cm of UO_2	60.48	25	13
Axial micro-heterogeneous, duplex 2.3 cm of ThO_2 with UO_2 and 1.1 cm of annular, graphite filled UO_2	57.06	19	7
Axial micro-heterogeneous, duplex –9.1 cm of ThO_2 with UO_2 core and 4.0 cm of annular, graphite filled UO_2	60.43	25	13
Axial micro-heterogeneous, 8.2 cm of ThO_2 and 5.0 cm of annular, voided UO_2	61.78	28	15
Radial micro-heterogeneous - ThO_2 and UO_2 pins in a 1x1 array	57.32	19	7

The major challenge of the axially microheterogeneous arrangements is to meet thermal hydraulic margins because of large local power peaking in the UO_2 driver section. The power peaking problem, illustrated in Figure 9, is a plot of the normalized power along the fuel rod axis in the region of the UO_2-ThO_2 interface of an axially microheterogeneous fuel rod at beginning-of-cycle (MacDonald et al. 2001b). Note that the power peaking is about a factor of 4.5 at the beginning of the fuel cycle for the worst case. Modified designs, which introduce some UO_2 in the ThO_2 section to improve power sharing, can significantly reduce this peaking to about 2.4, as shown in the lower line of Figure 9.

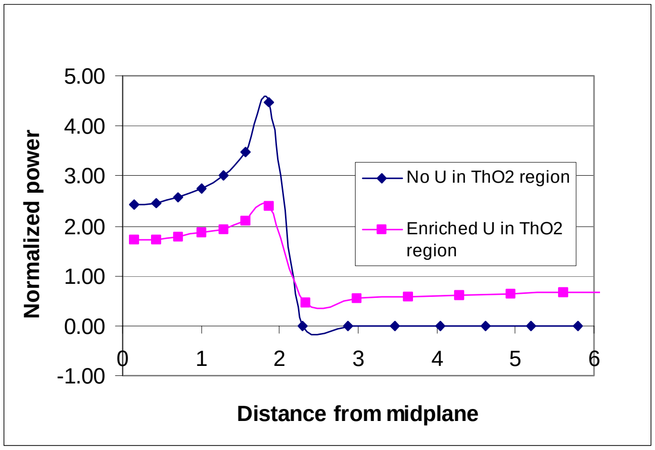


Figure 9. Normalized power at the UO_2/ThO_2 interface in axial microheterogeneous LWR fuel. The curve with enriched U is for the DuUAx4 design, i.e., the design with a small UO_2 core (19.5% enriched) in the ThO_2 region.

However, in spite of this peaking, the DNBR performance is predicted to be satisfactory due to the effects of the movement of the coolant from the low power to the high power regions of the axially micro-heterogeneous rods. (And then back to a low power region etc. which tends to average the coolant conditions in the driver and blanket sections of the axially micro-heterogeneous rods.) DNBR testing would be required to verify this assumed behavior prior to commercial reactor use. Use of annular fuel in the driver region significantly reduces the peak fuel temperatures, which then remain below the melting point of the UO_2. Nevertheless, the large temperature gradients raise other concerns, such as difficulties to satisfy LOCA constraints, hydriding of cladding, excessive gas release, and pellet/cladding mechanical interactions in the driver section. Significant irradiation testing of the thoria-urania fuel would be required to confirm the mechanical and fuel performance capability.

W8.2.e. Metal-Matrix Thoria-Urania Dispersion Fuel

Metal-matrix thoria-urania dispersion nuclear fuels have potential for use in a once through, high-burnup, high power, proliferation-resistant fuel cycle. The fuel is composed of a fine dispersion of thoria-urania micro-spheres in a zirconium metal matrix. About 50% of the oxide is thoria and about 50% is urania. The oxide fuel to metal matrix ratio is also about 1 to 1. The uranium enrichment is about 19.5%.

The pure zirconium matrix provides fuel and fission product containment, high thermal conductivity, and superior corrosion resistance during long reactor service and also during waste storage. The thermal conductivity of the metal matrix greatly enhances heat removal; thus the centerline fuel temperature will be significantly lower than that of a monolithic ceramic fuel pin. This latter point is important because the lower overall fuel temperature reduces the performance-limiting impact of fission product migration, fuel swelling, and other in-reactor phenomena. This can allow higher fuel ratings and fuel surface temperatures for use in supercritical water-cooled reactors and other advanced Generation IV reactors.

The potential benefits that may be gained with this proposed fuel form include low fuel fabrication costs due to the production of long length rods by a metal drawing process, high actinide burnup, inherent proliferation resistance, improved irradiation stability due to low internal fuel temperatures and stored energy, and high waste stability. The potential for high actinide burnup exists because the buildup of the U–233 during irradiation of the Th-232 can significantly extend the thorium fuel residence time.

Proliferation resistance arises from the use of a mixed oxide fuel, which makes the direct chemical separation of pure U–233 and Pu–239 impossible without subsequent isotope separation. The direct chemical separation of pure U–233 is not possible because it is intimately mixed with the U–238 from the UO_2 feed. The direct chemical separation of the Pu–239, or even low-grade Pu–239, is complicated by a significant quantity of Pu–238 and other even numbered plutonium isotopes at high burnup.

As a once-through system, this fuel is designed to be disposed after irradiation without processing and without encapsulation. The zirconium alloy matrix, Zircaloy shell, and Zircaloy cladding combine to form an excellent waste containment system. An additional waste disposal benefit arises because ThO_2 and $(Th,U)O_2$ are known to be more stable than UO_2 in oxidizing environments because the thorium does not have higher valence states available for further oxidation.

W8.3 POTENTIAL FOR CONCEPT MEETING GENERATION IV GOALS

W8.3.a. Evaluation Against High Level Criteria

In the following subsections, the ALWRs with Thorium/Uranium Fuel concept set are assessed against the Generation IV goals. The advantages and/or disadvantages of this concept set are evaluated relative to a typical Generation III reactor with a once-through uranium fuel cycle. In those areas for which no appreciable differences can be identified between the concept set and the reference, the concept set is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and metrics by means of a label in parenthesis.

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

Thorium fuel cycles have the following advantages relative to the current LWR once-through uranium fuel cycle with respect to Sustainability-1:

- Thorium is about 3 times more abundant than uranium in the earth’s crust. When bred to fissile U-233, thorium releases about the same energy as uranium bred to Pu-239 (SU1-1).

- Thorium is produced as a byproduct of the recovery of titanium, zirconium, tin, and rare earths. Limited demand for thorium has resulted in a worldwide oversupply, significant quantities of thorium in storage, and relatively low material prices (SU1-1, SU1-2).

- If plutonium from spent LWR fuel or weapons is used as the fissile material in a thorium fuel cycle, no uranium mining is required (currently available depleted uranium can be used to denature the U-233, if necessary) (SU1-1, SU1-2).

- Fuel cycle sustainability can be obtained with a U-233/Th-232 fuel cycle in a LWBR (SU1-1, SU1-2).

And thorium fuel cycles have the following relative disadvantages with respect to Sustainability-1:

- Thorium ore has no fissile component, therefore, fissile plutonium or U-235 must be added in relatively concentrated amounts in the once-through thorium fuel cycle designs (uranium with about 20% U-235 is used in most thorium-uranium designs). This results in modest savings in uranium mining (15 to 25% less) for the heterogeneous designs compared with the all-uranium fuel cycle (SU1-1, SU1-2).

It is concluded that the thorium-uranium once-through fuel cycle is only slightly more effective than the all-uranium once-through fuel cycle (i.e., significant uranium mining is needed to obtain the U-235). The thorium-plutonium fuel cycle is relatively sustainable in the near future because of the current plentiful supply of thorium and the availability of plutonium from both spent LWR fuel and weapons for burning. The Shippingport light water reactor breeder and other high conversion concepts with U-233 recycle are highly sustainable in the long term.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

Thorium fuel cycles have the following advantages relative to the current LWR once-through uranium fuel cycle with respect to Sustainability-2:

- UO_2 can be oxidized further to U_4O_9, U_3O_8, and UO_3. When UO_2 is oxidized to U_3O_8 a 30% volume increase occurs along with grain boundary separation and powdering of the fuel. This process releases most of the fission products trapped at grain boundaries and allows the fuel to easily dissolve in water. ThO_2 is the highest oxide of thorium and does not depart significantly from its stoichiometric composition when exposed to air or water at temperatures up to 2000°K (SU2-2, SU2-3).

- Mixed ThO_2-UO_2 fuel also appears to be a much better waste form than conventional UO_2 fuel, when the uranium content is below 50%. The lifetime of ThO_2-UO_2 fuel (exposed to wet air oxidation) appears to be on the order of millions of years rather than 100s of years for UO_2 (SU2-2, SU2-3).

- If U-233 recycle is used, the amount of high-level waste will be significantly reduced compared to any of the once-through fuel cycles (SU2-1). The long-term stability of the high level waste (primarily the removed fission products) will need to be determined.

- If the metal-matrix thoria-urania dispersion nuclear fuel form is used in a once-through cycle, the high-level waste material will be encapsulated in a dense, corrosion-resistant matrix that will enable secure disposal without additional containment. The zirconium matrix should enhance the inherent long-term stability of the $(Th,U)O_2$.

And thorium fuel cycles have the following relative disadvantages with respect to Sustainability-2:

- Regarding the Radkowsky once-through thorium fuel cycle, the mixed thoria-urania blanket rods are significantly more durable than UO_2 fuel as mentioned above, but metallic seed fuel rods are somewhat less durable than UO_2 because uranium metal will react with air and water (SU2-2, SU2 3).

Overall, the thorium fuels appear to be a significantly better long term waste than urania fuels.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

An effective international program currently safeguards the fissile materials within the civilian power programs in nearly all countries. However, it would be beneficial if future nuclear fuel cycles and nuclear safeguards systems, in combination, could provide an even higher degree of resistance to nuclear material proliferation. Specifically, advanced nuclear fuel cycles designed with intrinsic barriers may be more viable and effective over long periods of time than excessive reliance on extrinsic barriers. Nuclear fuel cycles that discharge a reduced quantity of weapons-usable material and a highly unattractive isotopic mix, and are transparent and inaccessible cannot be easily circumvented with changing political priorities.

Thorium fuel cycles have the following advantages relative to the current LWR once-through uranium fuel cycle with respect to Sustainability-3:

- The high burnup thoria-urania once-through fuel cycles (homogeneous, metal-matrix, or Radkowsky) will produce 3 to 6 times less separable weapons material (plutonium) per kW-hr than the current all-uranium fuel cycle because most of the fertile material will be thorium and the U-233 fissile material produced from the thorium can be denatured with U-238 (SU3-2).

- The plutonium in spent thoria-urania fuel will be much less desirable for use in weapons than the plutonium in the spent fuel from the current all-uranium fuel cycle (SU3-1). Thoria-urania fuel that is taken to high burnups contains a relatively small fraction of Pu-239 and relatively large fractions of Pu-238 and Pu-242. The even numbered plutonium isotopes in LWR spent fuel release spontaneous neutrons that significantly decrease the probable yield of a nuclear weapon. Pu-238 also releases significant heat that makes design and fabrication of a weapon difficult.

And thorium fuel cycles have the following relative disadvantages with respect to Sustainability-3:

- If a light water breeder reactor thorium fuel cycle with recycle of the U-233 is used, the fuel cycle will be less weapons material proliferation resistant than the current all-uranium once-through fuel cycle because of the separation of the U-233 from the thorium (SU3-1, SU3-2). (The critical mass for a U-233 weapon is about the same as for a Pu-239 weapon.)

- The once-through thorium fuel cycles require the use of uranium enriched to about 20% U-235 or plutonium fissile material (SU3-2). It is much easier to get to weapons grade material from 20% enriched UO_2 than from low (5%) UO_2 (24 versus 69 SWU per kilogram of 93% U-235), if enrichment facilities are available and misused.

Overall, it is concluded that the thorium once-through fuel cycles are significantly more nuclear weapons material proliferation resistant than the current all-uranium fuel cycles used in LWRs. The U-233 recycle fuel cycle is less weapons material proliferation resistant than the current all-uranium once-through fuel cycle currently used in LWRs.

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

Thoria fuel has somewhat different properties than urania fuel. These differences include (Belle and Berman 1984, Goldberg 1978, MacDonald et al. 2001):

- A slightly higher decay heat

- A higher thermal conductivity at normal reactor operating temperatures and a lower thermal conductivity at very high temperatures
- A slightly higher fission gas production per fission, but possibly a lower rate of release of fission gases
- A higher melting temperature
- Less plutonium buildup near the surfaces of the fuel pellets
- Less reactivity swing during the fuel cycle
- More negative moderator temperature coefficient
- More negative Doppler coefficient.

The Radkowsky once-through seed and blanket thorium fuel cycle proposes the use of annular metal or oxide seed fuel rods and metal fuel may swell, release fission gases, and possibly react with the water coolant in the case of a cladding breach. Annular fuel rods are more difficult to fabricate than cylindrical fuel rods. Also, the micro-heterogeneous oxide designs have a number of safety and reliability issues associated with their high power peaking and fuel temperatures in the driver or seed regions. And the U-233/Th-232 LWBR requires an extremely complex core design.

Therefore, the thorium fuel cycles have the following advantages relative to the current LWR once-through uranium fuel cycle with respect to Safety and Reliability-1:

- During normal operation ThO_2 and mixed ThO_2-UO_2 fuel will operate with somewhat lower fuel temperatures and internal gas pressures than UO_2 fuel at corresponding powers and burnups (SR1-3).

- The core will have a lower reactivity swing, more negative moderator temperature coefficient, and more control rod worth (SR1-3). These are particularly helpful if the cores are used to burn plutonium.

- The metal-matrix thoria-urania dispersion nuclear fuel design has a number of positive safety features resulting from the improved thermal conductivity of the fuel form. These include (a) a lower internal fuel temperature during steady-state operation, which mitigates swelling and other fuel performance issues, (b) reduced stored energy in the fuel in accident or rapid shutdown scenarios, and (c) a strongly negative void coefficient that enables consideration of advanced reactor concepts.

And thorium fuel cycles have the following relative disadvantages with respect to Safety and Reliability-1:

- The microheterogeneous oxide fuel has significantly higher power peaking (for example at the thoria-urania interfaces in the axial microheterogeneous rods) and fuel centerline temperatures in the driver regions than normal UO_2 fuel (SR1-3).

- The annular fuel rod design proposed for the Radkowsky seed rods may be more susceptible to end-plug welding defects (SR1-1, SR1-3).

- The seed metal fuel proposed for one of the once-through seed and blanket design options may be susceptible to excessive irradiation induced swelling and/or fission gas release. The metal seed fuel option may also be susceptible to more deterioration should there be a cladding defect than UO_2 fuel. More research is needed to define the irradiation behavior of that fuel. (SR1-1, SR1-3)

- The complex cores associated with the U-233/Th-232 fueled LWBR require additional quality assurance (SR1-3).

- All thorium-fueled cores will have a slightly smaller effective delayed neutron fraction because of the much smaller delayed neutron yield from U-233 fission than from U-235 fission. A smaller β_{eff} may lead to stricter requirements on the reactor control system and thus complicate the design.

Overall, it is concluded that the mixed thoria-urania fuel will have about the same reliability and safety as the current all-uranium fuel used in LWRs. However, the metal fuel and the micro-heterogeneous oxide fuel may have a lower reliability than the current UO_2 fuel.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

Again, thoria fuel has somewhat different properties than urania fuel. These differences include:

- A slightly higher decay heat

- A higher thermal conductivity at normal reactor operating temperatures and a lower thermal conductivity at very high temperatures

- A slightly higher fission gas production per fission, but possibly a lower rate of release of fission gases

- A higher melting temperature

- Less plutonium buildup near the surfaces of the fuel pellets

- Less reactivity swing during the fuel cycle

- More negative moderator temperature coefficient

- More negative Doppler coefficient.

Therefore, the thorium fuel cycles have the following advantages relative to the current LWR once-through uranium fuel cycle with respect to Safety and Reliability-2:

- During an accident such as a large break loss-of-coolant accident (LOCA), ThO_2 and mixed ThO_2-UO_2 fuel will have less stored energy but a slightly higher internal heat generation rate than UO_2 fuel at similar power levels. Calculations have shown that the resulting behavior of thoria-urania fuel and all-uranium fuel during a large break LOCA is essentially the same (SR2-1, SR2-2).

- The thorium fuel cycles have a much lower reactivity swing and significantly more negative Doppler feedback than the current UO_2 fuel cycles (SR2-1, SR2-2). Therefore, the postulated control rod ejection accident will insert much less reactivity and probably do much less damage.

- As noted above, the metal-matrix thoria-urania dispersion nuclear fuel will have even less stored energy than the ThO_2 and mixed ThO_2-UO_2 fuels.

And thorium fuel cycles have the following relative disadvantages with respect to Safety and Reliability-2:

- The metal seed fuel used in one of the once-through seed and blanket designs may melt and chemically interact with the cladding during certain design bases accidents (SR2-2). Much more analysis and testing is needed to understand and document its behavior in a water-cooled reactor during postulated accidents.

- The driver portion of the micro-heterogeneous fuel will experience higher temperatures and possibly more damage than UO_2 fuel during certain design basis accidents (SR2-1).

- In an accident scenario, the metal-matrix thoria-urania dispersion nuclear fuel micro-spheres may interact chemically with the zirconium matrix, despite the inherently low centerline temperatures. Particle coating methods may mitigate this issue, but that would add to the fuel fabrication cost.

Overall, it is concluded that the mixed thoria-urania fuel will have about the same low likelihood and degree of reactor core damage during a design basis accident as the current all-uranium fuel used in LWRs. However, the metal fuel and the micro-heterogeneous oxide fuel may experience somewhat higher core damage during a postulated design basis accident than the current UO_2 fuel.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

The advantages and disadvantages are the same as for Safety and Reliability 2 discussed above. Overall, it is concluded that the mixed thoria-urania fuel will have the same accident consequences (and need for offsite emergency response) as the current all-uranium fuel used in LWRs. However, the metal fuel and the micro-heterogeneous oxide fuel may experience slightly more damage during a postulated design basis accident than the current UO_2 fuel and the consequences of such an accident may be greater (what that means to the offsite emergency response is unknown). However, these differences are expected to be small.

Economics–1. Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

Thorium fuel cycles have the following advantages relative to the current LWR once-through uranium fuel cycle with respect to Economics-1:

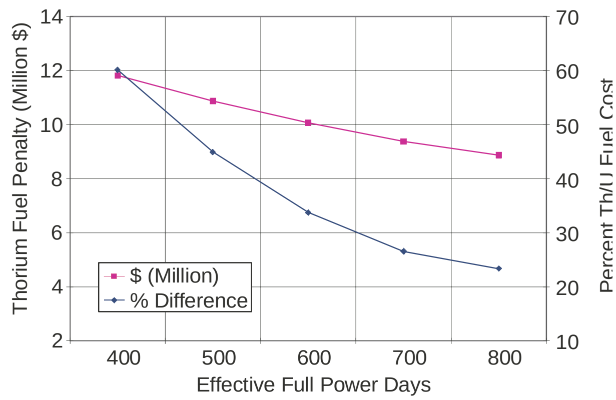
- Thorium is plentiful and relatively cheap (EC-3).

- The once-through seed and blanket designs may achieve the nonproliferation and waste disposal advantages of thorium utilization with little economic penalty on the fuel cycle costs and even modest savings for some designs (Galperin, 1999). Part of the savings are associated with the fact that less fuel will be manufactured because the blanket rods have such a long in-core residence time. (EC-3) [As discussed below, the major cost associated with the use of the once-through thorium cycles is the extra Separative Work Units (SWUs) associated with the use of relatively highly enriched uranium (uranium with about 20% U-235). These extra costs will occur regardless of which once-through thorium cycle is used, however, the seed and blanket designs provide the opportunity to compensate for those costs with lower fabrication costs.]

- In the long term, the worldwide uranium supplies will become tight and the yellow cake prices will rise, making the various thorium fuel cycles more attractive.

Thorium fuel cycles have the following relative disadvantages with respect to Economics 1:

- The homogeneously mixed once through thoria-urania designs, the micro-heterogeneous once-through oxide designs, the once-through metal-matrix thoria-urania dispersion nuclear fuel design, and the Radkowsky once-through seed and blanket thorium designs all require uranium enriched to 20% U-235. The SWUs required for 20% enriched material are considerably greater than for normal LWR fuel enriched to less than 5%. To the right is a plot of the additional costs (in millions of dollars per cycle) associated with the use of homogeneous thoria-urania fuel in a PWR versus fuel cycle length (MacDonald et al. 2001c). These costs are significant and extrapolation of the results gives no indication that further increases in burnup will make thoria-urania fuel economically competitive with the current UO_2 fuel used in LWRs (EC-3).



Thoria-urania cost penalty versus fuel cycle length.

- The costs associated with recycle of U-233 into a light water breeder reactor are unknown. However, U-233 enriched fuel fabrication must be done in a hot cell, so the fuel fabrication costs are likely to be much higher than fresh UO_2 fabrication costs and somewhat higher than MOX costs (EC-1). The costs of extracting the U-233 from the spent fuel using the Thorex process will likely be higher than the costs for extraction of plutonium from spent LWR fuel using the PUREX process (EC-3). (A pyrochemical processing alternative to Thorex that would reduce secondary waste volumes has been proposed based on molten fluoride salts; it is still in the developmental stage.) MOX recycle fuel in Europe is somewhat more expensive than fresh UO_2 fuel so we would expect that U-233/Th-232 recycle fuel will also be somewhat more expensive than all-uranium fuel.

- The once-through seed and blanket concepts require some additional fuel handling.

- The U-233/Th-232 light water breeder reactor fuel cycle will require a significant capital investment in recycling and fuel fabrication facilities (EC-1, EC-3).

Overall, we conclude that in today's market any of the proposed thorium fuel cycles may be more expensive than the current all-uranium fuel cycle. That situation may change as uranium supplies get used up and yellow cake prices rise.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

The power plant capital costs and risks of the thorium fuel cycles are approximately the same as for the current uranium fuel cycle used in ALWRs.

W8.3.b. Strengths and Weaknesses

Strengths of the thorium fuel cycles include:

- Very low plutonium production and very dirty plutonium isotopics (once-through fuel cycles).
- The light water breeder reactor will be fuel self-sufficient.
- Thorium is a plentiful and low cost material compared to uranium.
- Thorium dioxide is a very stable waste form.

The following are technical issues of the thorium fuel cycles to be addressed:

- Fabrication reliability and irradiation performance of annular metal seed fuel rods proposed for one of the once-through seed and blanket design options.
- Safety and reliability of the micro-heterogeneous thorium-uranium oxide fuel cycle designs.
- Safety, reliability, and irradiation performance of metal-matrix thoria-urania dispersion nuclear fuels.
- Costs associated with recycle of U-233 in a light water breeder reactor.

W8.4 TECHNICAL UNCERTAINTIES

W8.4.a. Research and Development Needs

The research projects discussed in Section 2 address the key technical issues identified thus far for these concepts. To summarize:

- The expected lower temperatures and gas release of the thoria-urania fuel need to be demonstrated with lead rod tests in commercial reactors and instrumented test reactor irradiations.
- The fabrication reliability and irradiation performance of the annular metal seed fuel rods proposed for the Radkowsky thorium fuel cycle concept needs extensive testing and demonstration.
- The fabrication reliability and irradiation performance of the metal-matrix thoria-urania dispersion nuclear fuels needs extensive testing and demonstration and continued fuel cycle development.
- The thermal-hydraulic performance of the various micro-heterogeneous thoria-urania designs needs analysis and testing.
- The design basis accident performance and safety of the annular metal seed fuel rods proposed for one of the once-through seed and blanket design options and the various micro-heterogeneous thoria-urania designs need extensive analysis and testing.
- Improved fuel rod cladding materials are needed for once-through seed and blanket thoria-urania blanket rods (nine-year irradiation) and the other various high burnup once-through designs.
- Simpler core designs are needed for the LWBR.

- The Thorex process and viable alternatives for recycling U-233 from spent U-233/Th-232 LWBR fuel needs further demonstration and costing.

- The fuel fabrication process for U-233/Th-232 fuel needs further development, demonstration, and costing.

W8.4.b. Institutional Issues - Licensability & Public Acceptance

No insurmountable licensability or public acceptance issues have been identified with this concept set. It is best characterized as an evolutionary fuel cycle design. A technically informed public should be receptive to the improved proliferation resistance and nuclear waste stability aspects of the once-through thorium fuel cycles. The fuel sustainability of the U-233/Th-232 LWBR fuel cycle will appeal to the public in the future when fuel resources are in greater demand and shorter supply.

The primary relative disadvantage of this concept set is the costs, which may be higher than the UO_2 fuel cycles currently used in LWRs.

W8.4.c. Timeline for Deployment

With strong research funding support, all of the thorium fuel cycle variations discussed in this appendix could be deployed by about 2015.

W8.5 INITIAL ASSESSMENT: OBSERVATIONS AND CONCLUSIONS

The cost penalties associated with the use of the various thorium fuel cycles discussed in this appendix will prevent their introduction in the U. S. and a number of western countries in the near future. However, certain countries with an abundant supply of thorium ore, and little uranium ore, will probably start using one or another of the various thorium fuel cycles earlier (e.g., India). As uranium supplies are depleted worldwide and as yellow cake prices rise the thorium fuel cycles will eventually become cost effective in all countries. The time frame for those changes in the economics may be longer than the time frame of this road map.

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W8.7. TOP-TIER SCREENING TABLE - THORIUM FUEL CYCLE

Summary Evaluation: X✗ Retain ____ Reject

Goal		- -	-	+	++	Comments
SU1	Fuel Utilization		E			-There is 3 times as much Th as U, however, once-through Th/U cycles use almost as much U as all-U cycles. -U-233/Th-232 LWBR fuel cycle is fuel-self-sufficient.
SU2	Nuclear Waste					-ThO_2/UO_2 is a much more stable waste form than UO_2. -U metal fuel is a somewhat less stable waste form than UO_2.
SU3	Proliferation Resistance					-The once-through Th fuel cycles generate significantly less Pu and very dirty Pu. -The U-233/Th-232 LWBR fuel cycles is less proliferation resistant than the reference
S&R1	Worker Safety and Reliability		E			-The mixed ThO_2-UO_2 fuel will have about the same reliability as UO_2. -The Radkowsky metal seed and micro-heterogeneous oxide fuel may have a lower reliability than UO_2 fuel.
S&R2	CDF		E			-The mixed ThO_2-UO_2 fuel will have about the same CDF as UO_2. -The Radkowsky metal seed and micro-heterogeneous oxide fuel may have a slightly higher CDF than UO_2 fuel.
S&R3	Mitigation		E			-Mitigation essentially the same as the all-uranium fuel cycle.

Goal		- -	-	+	++	Comments
E1	Life-Cycle Cost					-The SWU costs for all the once-through thorium fuel cycles and the recycling and fabrication costs for the U-233/Th-232 LWBR fuel cycle are currently higher than the reference. -The LWBR fuel cycle will require significant recycling facility and fuel fabrication plant capital costs. -The long-term thorium fuel cycle costs will be better than the reference when uranium supplies tighten.
E2	Financial Risk		E			-The power plant capital costs and finical risk of the thorium fuel cycles are similar to the all-U fuel cycle.

Appendix W9

Advanced Water-Cooled Reactors with Dry Recycling of Spent LWR Fuel

December 2002

CONTENTS

FIGURES

TABLES

ABSTRACT

Dry recycling of spent LWR fuel for use in either heavy water or light water reactors has been proposed for consideration as a Generation IV fuel cycle. The dry recycling technologies have been sufficiently studied to provide good confidence that they can be successfully deployed. Dry spent fuel recycling has the potential to meet several of the Generation IV goals and provides significant advantages in fuel utilization efficiency and reduction in nuclear waste production in comparison with the current once-through LWR fuel cycle. Overall, this fuel cycle concept has been assessed as worthy of retention for further consideration as a Generation IV option.

W9.1 INTRODUCTION

The proliferation-resistant, dry recycle of spent light water reactor (LWR) fuel into either heavy water reactors (HWRs) or LWRs addresses several of the Generation IV objectives. The application of this technology to either LWR/HWR or to LWR/LWR recycle has many similarities, although there are some important differences. The term "DUPIC," for "dry use of spent PWR fuel into CANDU" has been coined for the application of this technology for LWR-to-HWR recycle (Sullivan et al. 1999); the term "AIROX," for "Atomics International reduction oxidation" has been used for the process as applied to LWR/LWR recycle (Thomas 1993; Majumdar et al. 1992).

The technology may be particularly important and effective in addressing the accumulation of spent fuel in many countries, and in particular in the United States. Delays in developing geological repositories and hurdles in licensing either new spent fuel storage facilities, or expanded existing facilities, underscore the importance of recycling.

Other benefits of dry recycle are:

- Its high degree of proliferation resistance
- It is expected to be cheaper than conventional PUREX recycling and MOX fuel fabrication, and in the case of LWR/HWR recycle (DUPIC), it is expected to be cost effective compared to direct disposal
- It can effectively utilize ex-weapons fissile material (either Pu or high enriched uranium [HEU])
- The DUPIC cycle would significantly reduce uranium requirements compared to the once-through LWR fuel cycle
- It would reduce the heat load and cost of spent fuel disposal in a geological repository.

Dry-recycle could be utilized as a fuel cycle in Generation II reactors, in advanced Generation III reactors, and in next Generation IV reactors. The benefits are complementary to those of the Generation IV water reactor systems.

W9.2 CONCEPT DESCRIPTION

The developers of the concepts primarily wrote the concept summaries reported below. They have been edited for style and brevity. Some of their statements may not reflect the judgment of the Technical Working Group, which is reported instead in Section 3 of this appendix.

Spent PWR fuel nominally has a fissile content of ~0.9% ^{235}U, and 0.6% ^{239}Pu. This compares with the fissile content for fresh fuel of about 4% and 0.7% in current LWRs and CANDU reactors, respectively. In LWR/HWR recycle (DUPIC), the dry recycle involves a thermal/mechanical processing of the spent LWR fuel, to make new CANDU fuel, without the need for adding additional fissile material; in the case of LWR/LWR recycle, additional fissile material must be added to the recycled fuel powder. In both cases, there is no separation of uranium and plutonium, although one could also consider removal of rare earth, neutron absorbing fission products, to improve the burnup and fuel cycle economics.

In the DUPIC fuel cycle (see Figure 1), spent LWR fuel assemblies would be transported to the DUPIC fuel fabrication facility, where the fuel elements would be removed from the LWR fuel assembly, and the cladding removed. Several processes are feasible for de-cladding, including oxidation of the pellets, in which the volume expansion of the fuel pellets from UO_2 to U_3O_8 would rupture the clad.

The heart of the process is a series of oxidation ($UO_2 \rightarrow U_3O_8$ at ~400°C) and reduction ($U_3O_8 \rightarrow UO_2$ at ~600°C) steps, typically three cycles, which reduce the pellets to a fine powder (see Figure 2). The powder would then be milled to ensure the sinterability of the powder. Since there is no need for additional fissile material in the case of DUPIC fuel, the rest of the processing would follow normal CANDU fuel fabrication, only being done remotely: powder would be pressed and sintered to form new CANDU fuel pellets, loaded into new Zircaloy fuel cladding, and welded into CANDU fuel bundles. The small (10-cm diameter, 50-cm long), light-weight (~20 kg), simple design (there are only 7 components in a 37-element bundle) of the CANDU fuel bundles would greatly simplify the remote fabrication, and would help reduce the cost of DUPIC fuel fabrication. The use of the advanced CANFLEX bundle recently demonstrated in a current CANDU reactor; increases operating margins for DUPIC fuel cycles (see Figure 3).

During the oxidation/reduction cycles, and during sintering, volatile and semi-volatile elements such as cesium, krypton, iodine and xenon are driven off and must be captured, immobilized, and disposed. All other fission products and transuranic elements remain in the recycled fuel. The starting and end products, as well as the fabrication process, are highly radioactive; the fresh and recycled fuels must be transported in shielded flasks, and the re-fabrication process must be performed remotely in a shielded facility.

If spent PWR fuel having a nominal burnup of 35 MWd/kg is recycled into DUPIC fuel, then an additional burnup of >15 MWd/kg would be obtained through irradiation in current CANDU reactors. The fissile content (^{235}U and fissile Pu) in the spent DUPIC fuel is low, and there is no incentive for further recycle. The spent DUPIC fuel would then be stored, prior to eventual disposal.

Extensive studies have been done on the implications of using DUPIC fuel in CANDU reactors. The flexibility afforded by on-line refueling means that DUPIC fuel can be accommodated in existing and advanced CANDU reactors, using a simple bi-directional, 2-bundle shift fuelling scheme. Bundle and channel powers would be within current limits. The different kinetics parameters (smaller delayed

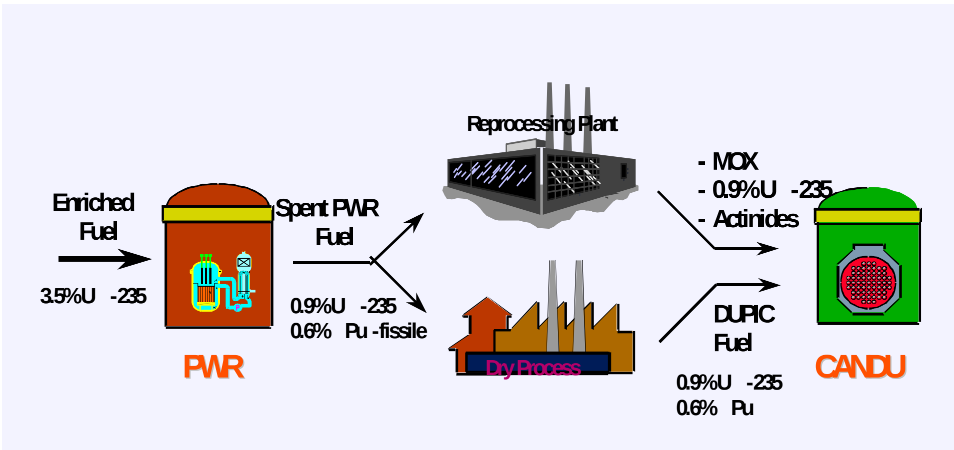

Figure 1. CANDU/PWR synergism.

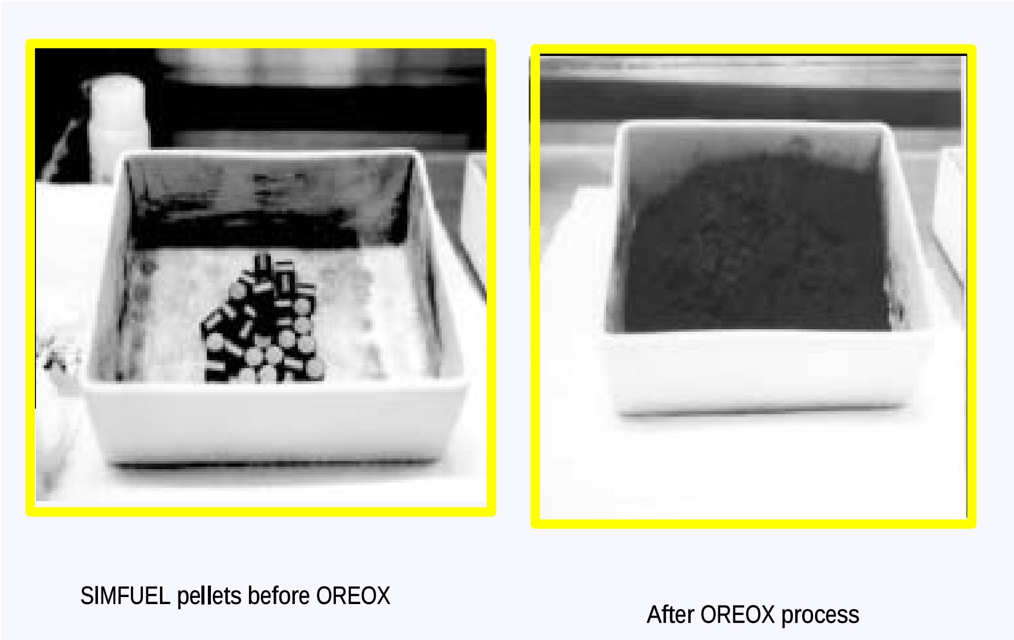

Figure 2. Oreox process.

Figure 3. CANFLEX fuel bundle after demonstration reactor irradiation at the Pt. Lepreau Nuclear Generating Station.

neutron fraction and neutron lifetime) can be accommodated in existing CANDU reactors by reducing void reactivity, through adding a small amount of neutron absorber to the central element in the DUPIC fuel bundles. (This would not be required in the next generation CANDU, where void reactivity with MOX fuel is negative). Minor refurbishment would be required to handle the radioactivity of the fresh DUPIC fuel. Two options have been considered for fresh DUPIC fuel handling: storing the fresh DUPIC fuel in the spent fuel bays, and "back-fuelling" from the bays into the fuelling machines and into the reactor, or building a new shielded fuel building, and providing shielding during fresh fuel loading. New CANDU reactors can be designed with the capability of handling radioactive DUPIC fuel from the start. Overall, DUPIC fuel can be accommodated with only minor changes in existing and future CANDU reactors.

The DUPIC program has been underway since 1991, with the participation of KAERI, AECL, and the United States (led by the Department of State, with participation by the Department of Energy (DOE), Arms Control and Disarmament Agency, and the Los Alamos National Laboratory [LANL]). The IAEA has participated in project review meetings since 1993. The first phase of the DUPIC program was a feasibility study that looked at a number of different methods of reconfiguring spent PWR fuel, to enable it to be burned in a CANDU reactor. Several options were judged to be technically feasible, and the current dry-recycle option was chosen for further study (Keil et al. 1992). It should be emphasized, however, that there are several other promising direct-use, dry-recycle options. The current phase started in 1994, and is focused on demonstrating the technical feasibility of the DUPIC cycle.

A major achievement in this program has been the dry-recycle of 3 kg of spent PWR fuel into CANDU DUPIC pellets, and the fabrication of 3 full-length CANDU DUPIC fuel elements, containing ~0.5 kg of DUPIC pellets in each element. Irradiation testing of these elements started 1999 March in the NRU reactor at AECL's Chalk River Laboratories. These elements have now reached a burnup >10 MWd/kg, and the first has been removed and is undergoing post-irradiation examination. Performance to date has been excellent. KAERI has also fabricated DUPIC fuel pellets that have undergone irradiation in a capsule in their HANARO reactor, and is currently fabricating several DUPIC elements from spent PWR fuel, for irradiation in NRU.

By now, many aspects of the DUPIC fuel cycle have been studied, confirming the technical feasibility. An important part of the program has been process and equipment development, including the strategy for ensuring the required degree of homogeneity when processing fuels having a wide variety of initial enrichments and discharge burnups. Technology has been assessed for fission gas capture and immobilization (Shin et al. 1999a; Shin et al. 1999b). The LANL in the United States has worked with KAERI in the development of unique safeguards technology (Hong et al. 1996; Lee et al. 1998). Detailed reactor physics, fuel management and safety studies have been carried out (Choi et al. 1997; Shen et al. 1998; Choi et al. 1998) including assessment of the impact of PWR fuel management strategy (initial enrichment, reload fraction, and discharge burnup) on DUPIC burnup (Shen et al. 1999). Fuel handling assessments have assessed options for handling the radioactive fresh DUPIC fuel at the station (Choi et al. 2001).

The extensive work done to date on the DUPIC fuel cycle has confirmed its technical viability and has considerably reduced the uncertainty in the various aspects of the technology. The flexibility of the CANDU reactor in accommodating a wide variety of fuels gives confidence in the use of DUPIC in CANDU.

The first steps of the AIROX process in the case of LWR/LWR recycle are similar to those for the DUPIC cycle, e.g., transport of the spent PWR fuel to the AIROX facility, de-cladding, and oxidation/reduction of the pellets. However, fissile material must be added to the recycled powder to meet the required fissile content (which is higher than the fissile content of the original enriched UO_2, to compensate for the reactivity load of the parasitic neutron absorbing isotopes in the recycled PWR spent fuel). Hence, a fraction of the recycled AIROXed powder must be replaced with virgin enriched powder. The fraction of the powder that can be recycled, or alternatively, the fraction of virgin enriched material that must be added, depends on the enrichment of the added virgin material. For instance (Zhao et al. 1999), if the enrichment of the original UO_2 is 4.5% ^{235}U, and the enrichment of the added UO_2 is 20%, then 22% of the recycled spent PWR powder must be replaced with new UO_2 enriched powder; if the enrichment of the added UO_2 is reduced to 10%, then 44% of the recycled powder must be replaced by new enriched UO_2. After the addition of the fresh enriched powder, the remaining processing steps are similar to conventional PWR fuel fabrication, only done remotely: the powder is pressed and sintered into pellets, then loaded into new fuel sheaths, and assembled into PWR fuel assemblies. While the remote fabrication would be considerably more complex than for CANDU fuel, the experience from PWR MOX fuel fabrication shows that this is feasible.

Since the fissile content of the spent, recycled LWR fuel is higher than in the original spent LWR fuel, there is incentive for subsequent recycling, either with re-enrichment for recycle in a LWR, or as-is for recycle in CANDU.

In the case of using dry-recycled fuel in an LWR, studies have shown that the core behaviour is intermediate between a full MOX core, and a high-burnup UO_2 core (Jahshan, S.M. et al. 1994). Hence, it would be anticipated that those reactors that can utilize a 1/3 MOX core, could also accommodate a 1/3 core of dry-recycled fuel. Similarly, those reactors that can accommodate a full-core of MOX, could accommodate a full core of dry-recycled fuel. Similar to CANDU, the fresh, radioactive recycled fuel could be stored in the spent fuel bays, and loaded into the reactor remotely (just as the spent fuel is removed from the reactor to the spent fuel bays.)

The study of LWR/LWR dry-recycle of spent fuel was initiated by Atomics International between 1959 and 1965. The characteristics of successive oxidation/reduction cycles were studied using un-irradiated UO_2 pellets containing oxides of stable fission product isotopes (SIMFUEL). Oxidative de-cladding was demonstrated at this time. The AIROX process was applied to a small quantity of spent fuel with burnups to 21 MWd/kg. Three stainless steel, 20-cm long irradiation capsules were used to irradiate pellets fabricated from AIROXed spent fuel (to which enriched UO_2 was added) to an additional burnup of 10 MWd/kg. These pellets were again remotely AIROX-processed with the view towards a second recycle, which did not take place as the program was terminated. The AIROX cycle was more recently assessed in the early ‘90’s (Thomas 1993; Majumdar et al. 1992).

It is noted that there is a synergism between LWR/HWR recycle (DUPIC), and LWR/LWR recycle. There is firstly a similarity in the recycle technology. Secondly, since LWR/LWR recycle cannot make use of all the spent LWR fuel, the AIROXed powder that cannot be recycled in a PWR could be recycled as-is as DUPIC fuel in a CANDU reactor. And finally, fuel that is once-recycled in an LWR would have an even higher fissile content than the original spent LWR fuel, and could be very efficiently recycled in CANDU as DUPIC fuel, without re-enrichment.

Another observation is made regarding the source of fissile material required for the AIROX-ed fuel in LWR/LWR recycle. Rather than enriched UO_2, weapons-derived fissile material could be considered (HEU or weapons-Pu). A much smaller amount of added fissile material would be required in the initial recycle, increasing the fraction of the spent LWR fuel that could be recycled. Since the cost of enriched UO_2 is a significant fraction of the recycled PWR fuel cost, depending on the price of the weapons-derived material, its use could substantially improve the economics of dry recycle. Moreover, its use in dry recycle would be an effective disposition option. A small amount of ex-weapons fissile material could also be added to the DUPIC powder, to substantially increase the burnup in a CANDU reactor, thereby reducing fuel cycle costs.

W9.3 POTENTIAL OF THE CONCEPT FOR MEETING THE GENERATION IV GOALS

In the following sections, the Advanced Water-Cooled-Reactors with Dry Recycling of Spent LWR Fuel concept set is assessed against the Generation IV goals. The advantages and/or disadvantages of the concept set are evaluated relative to a typical Generation III reactor with a once-through uranium fuel cycle, which serves as the reference system. In those areas for which no appreciable differences can be identified between the Dry Recycle concept set and the reference, the analyzed concept is rated E (i.e., Equivalent) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and by means of a label in parenthesis.

W9.3.a Evaluation against Criteria/Metrics

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

The dry recycle concept exhibits the following advantages in the area of natural resource utilization:

- In the DUPIC cycle, spent PWR fuel is dry-recycled directly into a CANDU reactor. Compared to the once-through PWR fuel cycle, this results in a reduction in natural uranium requirements of about 30% (Boczar et al. 1996). Selectively removing rare earth, neutron absorbing fission products would increase this to ~40% (equivalent to the so-called “TANDEM” fuel cycle). The fissile content in the spent CANDU DUPIC fuel is at tails level, and no further recycle is needed. This cycle has high fuel utilization, both in terms of efficiency of mined uranium, and efficiency of extracting energy from fissile material. (SU1-1)

- The use of either plutonium or high-enriched uranium (HEU) from dismantled weapons could significantly reduce the natural uranium requirements, in both the DUPIC cycle (a small addition of fissile material would increase burnup) and in the LWR/LWR recycle (by providing an alternate source of enrichment for blending with the spent fuel). Hence, with both options there is significant potential for significant improvement in fuel utilization. (SU1-1)

- The reduction in natural uranium requirements will also reduce mine tailings. The other major fuel cycle impact on the environment is from the dry-recycle processing and fuel fabrication facility. The absence of liquid waste from this facility will reduce the overall environmental impact of the fuel cycle. (SU1-2)

The dry recycle concept exhibits the following disadvantage in the area of natural resource utilization:

- In the LWR/LWR recycle, not all of the spent PWR fuel can be recycled after the AIROX process. Additional fresh enriched fuel is required to increase the fissile content to that required to achieve the target burnup. The higher the added fissile content, the larger the fraction of the spent LWR fuel that can be recycled. Typically, enrichments between 10% and 20% are considered in the fresh enriched uranium that is mixed with the recycled AIROXed spent PWR fuel. This higher enriched material must be extracted from additional mined natural uranium. The extra uranium offsets the savings in uranium resulting from recycling the fissile material contained in the spent PWR fuel. Typically, the total uranium utilization in LWR/LWR dry-recycle is about the same (or slightly worse) compared to the once-through LWR fuel cycle (Zhao, X. et al. 1999). (SU1-1)

The dry recycling of spent LWR fuel is assessed as better than the reference ALWR once-through fuel cycle when the material is used in a CANDU reactor. The LWR/LWR recycle is about the same or slightly worse than the reference.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

The dry recycle concept exhibits the following advantages in the area of waste minimization:

- Both concepts achieve a large reduction in high-level waste (mainly spent fuel). In an integrated system of LWR and CANDU reactors, the DUPIC fuel cycle would result in a 3-fold reduction in the quantity of spent fuel per unit energy, compared to direct spent fuel disposal in a dual system with CANDU reactors and once-through LWRs. The DUPIC system results in a 30% reduction in spent fuel relative to LWR fuelling alone (Sullivan et al. 1999; Boczar et al. 1996). LWR/LWR dry recycle can reduce spent fuel volumes by 30-50% (Thomas 1993; Feinroth 1998; Zhao et al. 1999). (SU2-1)

- There is not only a reduction in the volume of spent fuel to be disposed, but also in the heat load imposed on the repository (which impacts its size and cost), per unit electricity produced. In fact, the decay heat of the spent DUPIC fuel differs little from the decay heat of the spent LWR fuel from which it was derived, even though an extra 50% energy is derived from the fuel (Baumgartner et al. 1998; Ko et al. 2001). (SU2-1)

- The length of societal responsibility of the proposed fuel cycles will be similar to that of the reference once-through fuel cycles. However, the DUPIC cycle does reduce the long-term radiotoxicity of the spent fuel (Ko et al. 2001) because of the softer CANDU neutron spectrum, and the resultant destruction of certain actinides (which also results in a lower decay heat burden in the spent fuel). With LWR/LWR recycle, there is a small reduction in both the heat-load, and the long-lived radiological burden with spent PWR fuel that has been twice AIROX-recycled (Kuan et al. 1993). (SU2-3)

The dry recycle concept exhibits the following disadvantages in the area of waste minimization:

- The dry fuel recycling will generate a stream of volatile and semi-volatile fission products that must be collected, immobilized and packaged for storage and ultimate disposal. This is a disadvantage compared to the once-through LWR fuel cycle in which these radionuclides are kept immobilized in the ceramic UO_2 fuel matrix. (SU2-1)

- The dry-recycle processing facility must be designed to ensure low, even negligible, environmental discharges during normal operation. Once-through fuel cycles only require control of environmental emissions from the enrichment plant. (SU2-2)

The dry recycling of spent LWR fuel is assessed as better than the reference ALWR fuel cycle.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

The dry recycle concept exhibits the following advantages in the area of proliferation resistance:

- There is no purposeful separation of isotopes, nor can the processes be easily tampered with to effect such a separation and, therefore, the proliferation barriers that are present in spent fuel are also present in the dry-recycled fuel.

- The fuel processing does not involve any wet chemistry; only dry thermal–mechanical processes are employed. With no selective separation, the plutonium concentration remains dilute throughout the entire fabrication process, making it much more difficult to remove a significant quantity.

- All stages of the fabrication process, as well as the final fuel bundles or assemblies themselves, are highly radioactive, making physical access to the material, and its removal, extremely difficult. All processing and handling must be done in a shielded facility, this will result in highly automated processes with the inherent abilities to track movements and maintain fissile material inventory control.

- The specific radioactivity provides an easily detected “signature” of the material, making removal of material easy to detect.

- The processing facility is entirely self-contained: spent LWR fuel is an input to the facility, and finished CANDU DUPIC fuel bundles or LWR assemblies the product; there is no transport of intermediate products.

- Transportation of the spent PWR fuel into the processing facility and of new fuel to the reactor involve highly radioactive materials.

- The DUPIC option results in burning the ^{235}U isotope in the spent LWR fuel down to tails levels, as well as degradation of the plutonium vector (producing increasing amounts of higher-mass Pu isotopes), and consumption of plutonium, which provides another proliferation benefit. Successive recycles in the LWR will also degrade the plutonium vector. Increasing amounts of the ^{242}Pu isotope, as a result of multiple dry-recycles in a LWR, with its high spontaneous fission rate and high heat production is also a proliferation benefit. (SU3-1)

- The use of ex-weapons fissile material in either the DUPIC fuel, or in the AIROXed recycled LWR fuel, offers a means of dispositioning that material, while improving the economics of the dry-recycle options. (SU3-3)

The dry recycle concept exhibits the following disadvantages in the area of proliferation resistance:

- The required addition of enriched uranium (<20% 235U) in the AIROX process introduces a small additional proliferation risk compared to the once-through LWR fuel cycle because of the higher enrichment (easier to divert small volumes of material, easier to further enrich the material). (SU3-1)

- Recycle of spent fuel in CANDU reactors increases the levels of safeguards required to monitor and track the fuel bundle transfers with on-line re-fuelling, compared to spent PWR fuel storage or disposal. This disadvantage is offset by the fact the recycled spent fuel is inherently less attractive for diversion than natural uranium fuel. Special techniques have been developed for material accounting in DUPIC (Hong et al. 1996; Lee et al 1998). These can also be applied in the AIROX process, with LWR/LWR recycle. (SU3-2)

- In the LWR/LWR recycle, not all of the powder can be recycled (since some of it must be replaced with virgin enriched material). After a number of recycles, the material may have an excessive actinide burden. That material could be recycled as DUPIC fuel and the fissile content burned to tails levels. (SU3-3)

The dry recycling of spent LWR fuel is assessed as about the same as the reference ALWR fuel cycle.

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

The dry recycle concept exhibits the following advantages in the area of safety and reliability under normal operating conditions:

- The use of dry-recycled fuel should in principle not affect the reliability of the reactor, once the fuel has been fully qualified. Whether LWR or CANDU fuel, the fuel qualification will ensure that it meets requirements and performs well. The risk of fuel failures should not increase above the already low incidence in either LWR or HWR. For both recycled LWR fuel and CANDU DUPIC fuel, the fuel would perform within the current operating envelope (although both fuels would go to higher effective burnups). The on-line refueling in CANDU would enable the prompt removal of isolated defects without impacting on reactor operation. On the other hand, the proposed fuel cycle would not increase reactor reliability. (SR1-2)

The dry recycle concept exhibits the following disadvantages in the area of safety and reliability under normal operating conditions:

- It can be anticipated that the transport of spent LWR fuel to the recycle fuel fabrication facility, the fabrication of highly radioactive dry-recycled fuel, and its transport to the station, and fuel handing there will result in an increase in routine worker exposures, although these would be minimized through design and operating procedures to ALARA. There should be no increase in routine public exposures. (SR1-2)

- There are new risks to both workers and the public due to potential accidents at a recycle facility compared to the once-through LWR fuel cycle. (SR1-2)

- At this point, there is uncertainty in the performance of dry-recycled fuel, due to lack of irradiation experience. In both CANDU and LWR recycle, the gaseous, volatile, and semi-volatile fission products are removed from the spent LWR fuel during processing and fuel fabrication. The fission product free-inventory source term is “zeroed” at the start of the irradiation of the recycled material, and hence the free inventory that may be released during a reactor accident may be similar to that in current fuel. On the other hand, fission product release from the recycled material could be higher than from irradiated virgin UO_2, due to a degradation of thermal conductivity of the fuel from the presence of the fission products. The fission product release in any event would not be lower than for current fuel. (SR1-3)

The dry recycling of spent LWR fuel is assessed as moderately worse than the current LWR fuel cycle.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

The use of dry recycle fuel in the reference ALWRS or in CANDU reactors will have no significant effect on the likelihood or degree of reactor core damage.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

The use of dry recycle fuel in the reference ALWRS or in CANDU reactors will have no significant effect on the need for offsite emergency response.

Economics–1. Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

The dry recycle concept exhibits the following advantage in the area of operating costs:

- In the DUPIC fuel cycle, the cost of DUPIC fuel fabrication offsets the avoided costs of UO_2 for CANDU fuel fabrication and the costs for spent PWR fuel storage and disposal. There are additional costs in the DUPIC cycle such as the transportation of spent PWR fuel from a PWR power plant to the DUPIC fuel fabrication plant, and transportation of fresh DUPIC fuel from the DUPIC fuel fabrication plant to the CANDU power plant. Fairly detailed, (albeit preliminary) cost estimates of the DUPIC fuel cycle cost have been made (Choi, H. et al. 2001a; Choi, H. et al. 2001b; Ko, W.I. et al. 2001a; Ko, W.I. et al. 2001b), which indicate that within the uncertainties of the cost parameters, the DUPIC fuel cycle cost is lower than for conventional PUREX recycling (and MOX recycle) of PWR fuel, and is competitive compared to direct disposal. These cost analyses are necessarily preliminary, since more work is required to define the technical aspects of the DUPIC cycle. (EC-3)

The dry recycle concept exhibits the following disadvantages in the area of operating costs:

- The economics of the AIROX LWR/LWR recycle have also been recently assessed using nominal cost assumptions (Zhao, X. et al. 1999). The study looked at multiple AIROX recycle of spent LWR fuel, using either 10% enriched or 20% enriched uranium feed (the latter allowing twice as much spent fuel to be recycled). In both cases, the extra cost of the uranium enrichment (SWU) in the feed material, along with the higher AIROX fuel fabrication cost, more than offset the avoided spent LWR disposal costs. For these cost assumptions, the cost of AIROX recycle was slightly more than the cost of direct disposal. (EC-3)

- While the referenced DUPIC cost comparisons show similar economics to the once-through cycle there is significant uncertainty in the estimates.

- Implementation of the dry spent fuel recycle process requires substantial investment in the development and construction of the spent LWR fuel recycle facilities. The facilities will be expensive because of the need to contain volatile fission products, and remote handling. There will be a risk in the siting and licensing of the spent fuel recycle facility. (EC-1, EC-2)

The dry recycling of spent LWR fuel is assessed as ranging from moderately worse to moderately better than the reference ALWR once-through fuel cycle.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

The dry recycle concept exhibits the following advantages in the area of power plant capital costs and financial risk:

- Additional R&D is needed, however, the small, simple CANDU bundle design, the ability to recycle the spent PWR fuel directly into CANDU without adding additional fissile material, and the relatively low burnup of the DUPIC fuel, simplifies the LWR/HWR processes and R&D requirements. The technical feasibility of the LWR/HWR fuel cycle is currently being demonstrated in a joint program with AECL, KAERI and the U.S. DOE.

- These fuel cycles can be implemented without the necessity of significant changes to the current CANDU and ALWR plant and reactor core designs. (EC-2)

The dry recycling of spent LWR fuel is assessed as equivalent to the current LWR fuel cycle.

W9.3.b Summary of Concept Potential

The potential of this concept is summarized below. Overall, the concept is a reasonable candidate for further consideration.

Table 1. Dry recycling of spent LWR fuel: concept strength and weakness.

Category	Strengths	Weaknesses
Sustainability	• Significant reduction in spent fuel disposal volume • With DUPIC, no additional fissile material is required; 30-40% reduction in uranium requirements; reduction in spent fuel heat load, and long term radiotoxicity • Synergistic with excess-weapons fissile material dispositioning • No liquid effluents	• With LWR/LWR recycle, need to add fissile material (not all spent fuel can be recycled); little impact on natural uranium requirements • Application of safeguards (material accountability) somewhat more challenging
Safety	• Safety features of Gen IV reactors not degraded through use of dry recycled fuel • Advanced fuel design features could be incorporated into CANDU DUPIC fuel, to enhance fuel performance during normal operation and postulated accidents	• Small increase in routine worker exposure likely • Much more irradiation experience with recycled fuel is required

Category	Strengths	Weaknesses
Economics	• Can be introduced into existing reactors with only minor changes to systems • Fuel cycle costs can be reduced using ex-weapons fissile material • Technical feasibility demonstrated, which reduces risk • With DUPIC, lower fuel cycle cost than recycling, and competitive fuel cycle costs compared to direct disposal; • In DUPIC fuel, small, simple, light CANDU bundle facilitates low cost remote fabrication	• With LWR/LWR recycle, slightly higher fuel cycle costs than once-through PWR due to adding high fissile content uranium

W9.4 TECHNICAL UNCERTAINTIES: R&D REQUIREMENTS

W9.4.a. Research & Development Needs

The work done to date on the dry recycle of spent LWR fuel has confirmed its technical viability. The R&D needs now center on furthering our understanding of dry-recycle technology to reduce uncertainties with regards to fuel fabrication requirements and costs, and fuel performance.

For the dry recycle of spent LWR fuel in CANDU reactors, more DUPIC fuel should be made from a variety of LWR spent fuel sources to understand and to optimize the fabrication parameters; the fuel should be irradiated under a variety of conditions (including power ramps) and then examined to understand its in-reactor performance. This will establish the sensitivity of the DUPIC fuel performance to its pellet characteristics and fabrication parameters.

It is also necessary to extend the fabrication processes and associated equipment from laboratory scale, to a pilot scale facility, and finally to a commercial scale facility. An important element of the fabrication process will be capture of the fission gases released during processing of the spent LWR fuel, and technologies need to be developed for both gas capture and immobilization.

Demonstration bundles should be fabricated and irradiated in CANDU power reactors. This work will also be necessary to reduce the uncertainty in DUPIC fuel cycle economics.

Continued development is required for safeguards technology, and more detailed analysis needs to continue on the in-reactor neutronics and safety, as well as fuel handling at the station.

The R&D requirements for LWR/LWR recycle using AIROX are similar to those for DUPIC fuel. Moreover, because of the commonality of the core technology, this technology can benefit from experience obtained with DUPIC fuel.

For LWR/LWR recycle, reference (Thomas 1993) identifies the technical gaps and R&D needs: material accountability of the spent fuel assemblies (an issue in common with DUPIC, for which considerable progress has been made which can also be applied to LWR-recycle [Hong et al. 1996; Lee et al. 1998]), de-cladding, oxidation/reduction on multiple-recycled material, ball-milling to appropriate size, and mixing of spent and virgin UO2, remote fabrication, and off-gas recovery and immobilization. Various paper studies have been conducted, including reactor physics assessments, and conceptual AIROX fuel fabrication facility design. The in-core neutronics performance of AIROX-ed fuel has been shown to lie within the neutronics performances of existing high burnup fuel, or equivalent MOX fuel (Jahshan et al. 1994).

The fuel performance demands on LWR/LWR recycled fuel are greater than for DUPIC fuel for the following reasons. With DUPIC fuel, the fissile content is burned down to tails level in one cycle in the CANDU reactor, and there is no need or incentive for further recycle. Roughly 50% more energy is extracted from the spent LWR fuel after recycling in CANDU (e.g., the equivalent LWR burnup increases by ~50%). With LWR/LWR recycle, the fuel is taken to double the initial PWR burnup on the first recycle, and equivalent to 3-times the LWR initial burnup on the second recycle (>100 MWd/kg). The reconstituted fuel can therefore be taken to a very high effective burnup. This aspect will require additional qualification, and can benefit from the considerable experience-base with high burnup fuel. Remote fuel fabrication for recycled LWR fuel will have additional challenges as well, due to the complexity and size of LWR fuel elements and bundles. Worldwide experience in remote MOX fuel fabrication can be applied.

More detailed reactor physics studies for AIROX-processed fuel are also required, for both existing and Generation IV LWRs. The effect of the presence of fission products and the larger amounts and variety of actinides on the reactivity worth of reactor control and safety needs to be analyzed.

Another area warranting further study is the synergism between LWR/HWR, and LWR/LWR recycle.

Table 2 summarizes the major R&D requirements for advancing the dry recycle technology. The development rating is given for a combination of cost and risk. Most of the activities are anticipated to be moderate owing to the cost of handling irradiated materials and performing test irradiations.

W9.4.b. Institutional Issues – Licensability and Public Acceptance

There are no significant licensing issues associated with this fuel cycle. The proposed dry recycling technologies have been sufficiently studied to provide good confidence that they can be successfully deployed. A successful development program would include qualification irradiations to demonstrate the performance of the advanced fuel to meet regulatory requirements. The fuel would be used in licensed reactor designs within an accepted licensing envelope for those designs, based on reactor physics, neutronics and safety assessments.

There are potential public acceptance issues. While the dry recycle process offers many benefits, including greater uranium utilization, improved actinide burning and lower spent fuel volumes for storage and disposal, there are features, many common to all recycling technologies, that may be viewed as unattractive. These include:

- Increased safeguards issues
- Additional fuel bundle transfers to be tracked for CANDU fuel
- A requirement for moderately enriched uranium (10 to 20%) for the LWR/LWR fuel cycle
- Generation of a new waste form from separated fission gases and attendant risks

Table 2. Dry Recycle Technology Development Requirements

Development Area	Requirement	Development Rating
DUPIC		
Fuel Pellet Design	Test irradiations	Moderate
Fission Gas Capture and Immobilization	Development of commercial scale processing technology	Moderate
Manufacturing Process	Scaling of equipment and processes	High
Demonstration Irradiation	Fuel qualification and economics confirmation	Moderate
Physics	Neutronics and physics assessments	Low
Safeguards	Accountability processes	Low
LWR/LWR Recycle		
Fuel Fabrication	Optimization of decladding, repeat oxidation, fuel mixing and remote assembly	Moderate
Manufacturing Process	Scaling of equipment and processes	High
Fission Gas Capture and Immobilization	Development of commercial scale processing technology	Moderate
Extended Burnup Performance	Test irradiations	Moderate
Demonstration Irradiation	Fuel qualification and economics confirmation	Moderate
Physics	Neutronics and physics assessments	Low
Safeguards	Accountability processes	Low

- Risks due to additional transportation of radioactive materials
- Risks due to potential recycling plant accidents
- Additional recycling plant worker risks.

Overall dry recycling is judged to have fewer issues than convention recycling, but more than the reference once-through fuel cycle.

Time Line for Deployment

The R&D required for deployment of the dry fuel recycle technology is relatively modest. Extensive work has been completed on the DUPIC fuel cycle and DUPIC test fuel pellets and elements have been manufactured and irradiated. The performance to date has been excellent. While further work is desirable and necessary to optimize the fuel performance and manufacturing parameters, the major step in deployment will be extension of the fabrication processes and equipment to commercial scale. This should be achievable within a period from 2010 to 2015.

The LWR/LWR dry recycle process is less mature and additional work is required to bring it to the same level an the DUPIC fuel cycle in terms of test element manufacture and irradiation. Additional technical requirements and higher costs for the assembly of longer LWR fuel elements could extend the time required to develop a commercial scale manufacturing capability for this recycle technology as well. The time for deployment of this fuel cycle is estimated to be about 2015 to 2020.

W9.5 STATEMENT OF OVERALL CONCEPT POTENTIAL VERSUS R&D RISK

The proposed dry recycling technologies have been sufficiently studied to provide good confidence that they can be successfully deployed. These technologies have significant potential benefits for reducing spent fuel volumes, increasing fuel utilization, reducing proliferation risk in recycle, and in enhancing long-term sustainability. Furthermore, they can be employed in both existing reactors, and in next generation reactors, complementing the benefits from those reactor designs. Therefore, it is recommended that the dry recycling concepts be retained for further consideration.

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W9.7 TOP-TIER SCREENING TABLE – ADVANCED WATER-COOLED REACTORS WITH DRY RECYCLE OF SPENT LWR FUEL

Summary Evaluation: __X__ Retain _____ Reject

Goal		--	-	+	++	Comments
SU-1	Fuel Utilization		E			-DUPIC has advantages in that 30% more energy is produced from spent fuel. -The LWR/LWR dry recycle has no advantages over the reference once-through fuel cycle
SU-2	Nuclear Waste					Both concepts reduce volumes of waste
SU-3	Proliferation Resistance		E			Better than conventional PUREX recycling & similar to the reference once-through ALWR fuel cycles
S&R-1	Safety and Reliability		E			Compared to the reference once-through fuel cycle, some safety concerns associated with fuel processing
S&R-2	CDF		E			
S&R-3	Mitigation					

Goal		--	-	+	++	Comments
E-1	Life-cycle cost					-DUPIC fuel cycle cost comparable to once-through -LWR/LWR likely higher due to "top-up"
E-2	Capital Costs & Financial Risk					

Appendix W10

Advanced Light Water Reactors with Plutonium and Minor Actinide Multi-Recycling Concept Set Report

December 2002

CONTENTS

FIGURES

TABLES

ABSTRACT

These are ALWRs (thermal spectrum reactors) that are designed to multi-recycle the plutonium and minor actinide fissile material. Plutonium and minor actinide multi-recycling in Generation-4 water-cooled-reactor nuclear energy systems has the potential to reduce high-level waste burdens, extend uranium resources, reduce enrichment requirements, and therefore improve the sustainability of nuclear power. The use of plutonium in LWR cores requires careful attention to the issues of maintaining criticality to high burnup, neutron energy spectrum hardening, control rod effectiveness, core transients, void reactivity coefficient, power peaking, and safeguards against diversion of fissile materials. Minor actinide recycling would be most effective with an improvement of the decontamination factor achieved during reprocessing to minimize the fraction of minor actinides that escape the cycle and go to waste disposal. Effective shielding or remote handling will be required for a minor actinide recycle fuel fabrication facility.

A number of fuel designs have been developed for plutonium and minor actinide multi-recycle, some of which are: MIX, CORAIL, and APA. The MIX concept uses a homogeneous mixture of oxides (UO_2 and PuO2) in each fuel rod. The CORAIL concept uses a heterogeneous arrangement of UO_2 rods and MOX rods, and the APA concept uses a heterogeneous arrangement of UO_2 rods and rods with PuO2 in an inert matrix.

Except for MIX, these core designs are mainly at early stages. Much additional R&D is needed on the details of the fuel assembly design, safety analyses, reprocessing, fuel fabrication, and cost estimates.

W10.1 INTRODUCTION

The incentives for recycling of plutonium and minor actinides include the desire to extend uranium resources, the need to reduce the radiotoxic inventory of materials going to waste disposal, and the desire to burn up surplus fissile materials from dismantled weapons. For example, it is estimated that there are about 50 tons of surplus weapons grade plutonium in the United States and 100 tons in Russia [Magill et al. 1997].

France has assessed plutonium and waste management for pressurized water reactors (PWRs) and fast breeder reactors (FBRs). Considerable R&D work has already been performed to improve the use of plutonium in PWRs. Currently the plutonium is put in PWR cores partially loaded with mixed oxide (MOX) fuel assemblies (Figure 1) only once (mono-recycling). European MOX irradiation experience extends to 52 GWd/t (rod average burnup) in commercial PWRs and 60 GWd/t in experimental assemblies, with good dimensional behavior, corrosion behavior, and properties similar to UO_2 [IAEA 1999].

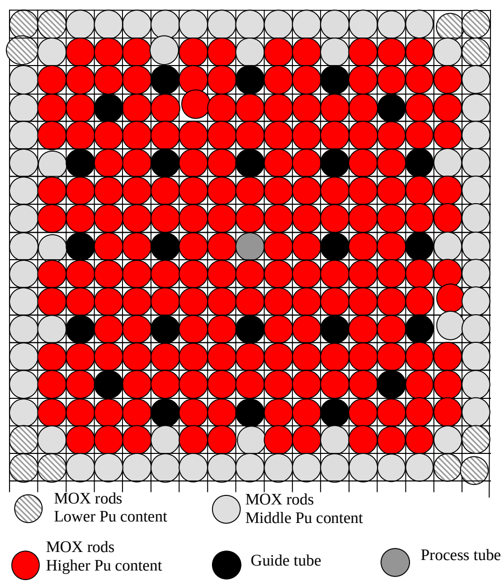

Figure 1. MOX fuel design.

The first French reactors (900 MWe), initially licensed to use enriched UO_2 fuel, were slightly adapted to accept plutonium. For more efficient use of plutonium in PWRs several fuel concepts are currently being examined. The objective of these innovative fuel concepts is to facilitate core management in a plutonium multi-recycling strategy and to increase fuel burn up performance, keeping safety margins the same as for current UO_2-fuelled PWRs. However, there are several issues that must be considered when planning to recycle plutonium and actinides in PWRs.

- Changes in the core reactivity (k) must be accommodated. This is taken into account by adjusting the quantity of fissile plutonium isotopes in the MOX fuel (such as by addition of surplus weapons grade plutonium) or by adjusting the ^{235}U enrichment. If the uranium from spent fuel is recycled, the buildup of ^{236}U (a strong neutron absorber) requires very high enrichments to sustain criticality in subsequent cycles [Waris and Sekimoto 2001]. As the quantity of even-numbered plutonium isotopes with low thermal fission cross-sections increases, the total quantity of plutonium must be increased to maintain the core critical. Alternatively, surplus weapons-grade plutonium or high enriched uranium (HEU) could be added to the MOX, if available, but the HEU could be expensive.

- The presence of plutonium hardens the neutron spectrum, which reduces the worth of control rods and soluble boron. The control problem can be alleviated by limiting the plutonium content, by redesigning the control rod assemblies, or by improved neutron moderation. In some French PWRs, the boron concentrations were increased and four-rod cluster control assemblies were added without significant economic penalties so that cores loaded up to 30% in MOX could be accepted. For the European Pressurized Reactor (EPR), these control system improvements were included in the design phase, allowing the partial loading of 50% MOX assemblies [Grouillier 2001].

- The safety margins change. Voiding of coolant channels also hardens the neutron spectrum. At high neutron energies all the isotopes of plutonium can undergo fission, which increases the reactivity. From a study of six widely different mixes of plutonium isotopes, it was found that the limiting total plutonium fractions (above which the void coefficient becomes positive) vary from 12.5% (with 90% fissile plutonium) to 15% (48% fissile plutonium) [Aniel 1997]. ^{239}Pu also has a much lower delayed neutron fraction (0.0021) than ^{235}U (0.0065), which makes kinetic control more challenging for cores with high plutonium content. To minimize control problems the MOX fuel fraction in the core is often limited to 30% in PWRs [Hesketh 1997].

- Change of power peaking. This can be resolved by adjusting the fuel loading pattern to minimize the hot channel factor. For example, in CORAIL assemblies the MOX rods are located at the periphery of each fuel assembly, where the thermal flux is the lowest.

- Fuel fabrication problems occur when significant quantities of actinides with high internal heat generation, gamma emission, and neutron emission are included. For example, ^{241}Pu (14.4 y) decays into ^{241}Am, which is a strong gamma emitter and neutron absorber, so the ^{241}Pu should be used within a few years to avoid pollution of the fissile material. ^{241}Am (433 y) also decays into ^{237}Np (2.14×10^6 y), which creates a waste disposal problem. Depending on mass and configuration, a concentration of 5% ^{238}Pu in fissile materials could increase their surface temperature up to about 875°C [Cochran and Tsoulfanidis 1990]. Recycling of Curium would increase the neutron emission rate by a factor of 100, which would require special shielding or remote handling [Renard 1995].

- Increase of the total quantity of plutonium in the system. If it is desired to minimize further production of plutonium in MOX-fueled reactors, then the PuO_2 fuel can be placed in an inert matrix, instead of in UO_2. Plutonium production can also be controlled by increasing discharge burnup, by adjusting the ratio of hydrogen to heavy metal, or by the addition of thorium fuel [Kazimi 2001]. (See also Appendix W8 on Thorium Fuel Cycles.)

- The radiotoxic inventory in the core increases with repeated recycling of plutonium and minor actinides. The radiotoxic inventory of nuclear materials can be effectively reduced only by irradiating them in a fast neutron spectrum. For example, this could be done in a fast reactor, in an accelerator driven system, or in a fusion reactor. Accelerator driven systems can produce very high neutron fluxes ($\sim 10^{16}$ n/cm^{2}s) but are energetically expensive and capital intensive [NEA 1998]. Fusion reactors could also produce high fluxes of 14 MeV neutrons, but reliable long-term operation at high power has not yet been demonstrated.

The philosophy in Russia is to use fast reactors for waste transmutation, instead of plutonium recycle in light water reactors (LWRs), for the following reasons [Mikhailov 1994]:

- The change of the LWR safety margins with use of plutonium fuel will make licensing difficult.

- One fast reactor can burn as much excess weapons plutonium as six LWRs.
- Recycle increases the fractions of 241Pu and 241Am, which exacerbates waste disposal problems.

W10.2 CONCEPT DESCRIPTION

Neutron thermalization in MOX-fuelled PWRs may be improved by limiting the plutonium mass or by increasing the moderation ratio (the ratio of hydrogen atoms to fuel atoms, which is typically about 3.4 in PWRs) [Kazimi, 2001].

W10.2.a. Mix

With homogeneous fuel we can limit the plutonium content in all assemblies and add ^{235}U to comply with fuel management constraints. This is the **MIX** concept [Delpech 1998], which uses all MOX rods, but with varying PuO_2 content in a standard PWR fuel assembly configuration. (Figure 1 shows one version of this concept.) This concept maintains the safety margins similar to those of the all-uranium core and also maintains the criticality by adjustment of the ^{235}U enrichment. The plutonium content in the various rods may vary from approximately 2 to 6%. Between 100 and 30% of the fleet PWRs could be involved with MIX loading. Feasibility studies have established the average plutonium content limit at 4% in the P4 type of French 1300 PWRs. A COGEMA study found that with multiple recycling, the MIX fuel and the standard UO_2 fuel would have comparable fuel costs if the MIX fuel fabrication cost did not exceed the UO_2 fuel fabrication cost by more than 400 $/kg [Durret et al. 1997]. A study of 5 recycles in highly moderated MOX cores found that plutonium consumption could be enhanced while satisfying the core nuclear and thermal design criteria [Iwata et al. 2000].

W10.2.b. CORAIL

The **CORAIL** concept uses a heterogeneous arrangement of MOX rods (PuO_2 in a depleted UO_2 matrix) and UO_2 rods in a fuel assembly [Youinou 2001]. This reduces the neutron spectrum hardening and the required enrichment relative to the MIX concept. There are several ways to distribute the two types of rods in the assembly. Figure 2 shows an example of the CORAIL concept with UO_2 rods surrounded by MOX rods. The plutonium content increases during subsequent recycling, but in less significant proportions than in the MIX concept, thanks to the presence of the ^{235}U-enriched UO_2 rods.

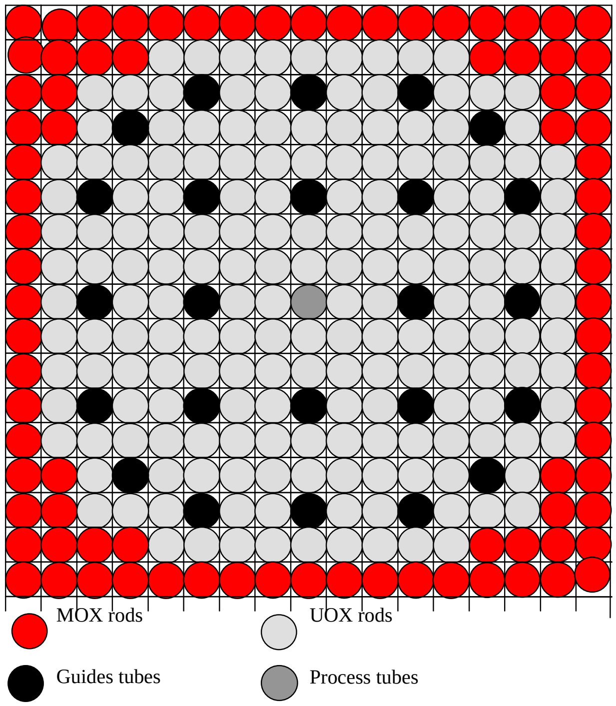

Figure 2. CORAIL fuel design.

The **DUPLEX** concept is similar to **CORAIL** but uses an inert matrix for the PuO_2 instead of UO_2 (Figure 3). DUPLEX is not discussed further here, because results are not yet available. (The name *duplex* is also associated with a different design concept for microheterogeneous fuel pellets with separate regions of ThO_2 and UO_2 [Figure 8 of Appendix W8].)

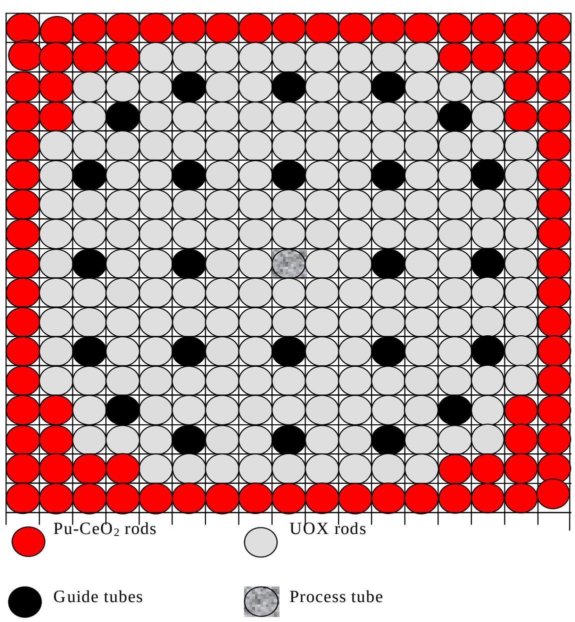

Figure 3. DUPLEX fuel design.

W10.2.c. APA

We can increase the moderation ratio by increasing the volume of water, as done in the Advanced Plutonium Assembly (**APA)** concept [Puill 1999], or by decreasing the fuel density. The APA assembly consists of a heterogeneous arrangement of PuO_2 in an inert matrix (CeO_2) surrounded by UO_2 rods. For example, an annular PuO_2 rod can replace 4 standard rods (Figure 4).

This annular fuel design facilitates enhanced spectrum thermalization, with a local moderation ratio of ~ 8. Other APA rod designs, such as small PuO_2 rods or cross-shaped PuO_2 rods, are under study. The APA concept could reduce the amount of TRU going to waste disposal from 13.4 tons/y (open cycle) to 3 tons/y [Golfier 2001].

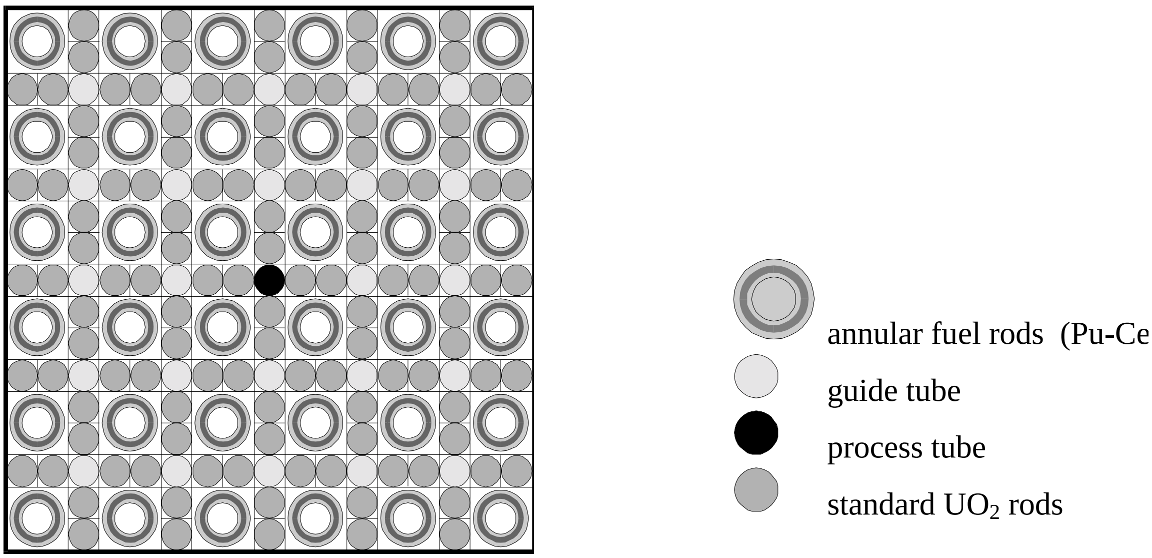

Figure 4. APA fuel design.

Table 1 shows the main characteristics of the MIX, CORAIL and APA concepts for plutonium in PWR multirecycling, compared to conventional UO_2 cores and 30% MOX fuel loadings for once-through cycling of plutonium in PWRs. One notes that:

- The APA assembly concept comprises the lowest number of plutonium rods (a factor of 7 with respect to MIX and a factor of 2 with respect to CORAIL). The total number of rods in the APA assembly is also lower than that of the other concepts by a factor of 1.7.

- The plutonium coming from the APA fuels is of a lower quality than that coming from MIX or CORAIL fuels (21% of fissile plutonium against 48% for CORAIL and 54% for MIX).

- The APA concept results in a quantity of minor actinides in the assembly of 7.9%, which is much larger than those in the CORAIL concept (2.8%) and MIX concept (2.4%).

Table 1. Main characteristics of various concepts for plutonium recycling in PWRs.

	UO_2 open cycle	MOX mono-recycling	MIX Equilibrium	CORAIL 7th Recycle	APA 4th Recycle
Burn up (GWd/t)	55	*55*	55	*45*	90
Fuel management (batches, days between refueling)	6, 270	6, 270	6, 270	3, 440	5, 280
UO_2 rod :					
^{235}U enrichment (%)	4.5	-	-	4.80	3.2
Rod with plutonium :					
^{235}U enrichment (%)	-	0.25	3.8	0.25	-
Plutonium content (%)	-	10.4	2.0	8	100
Rod number with plutonium :	-				
UO_2		-	-	180	120
MOX		264	264	84	-
Inert plutonium		-	-	-	36
	UO_2 open cycle	MOX mono-recycling	MIX Equilibrium	CORAIL 7th Recycle	APA 4th Recycle
Subassembly mass (Kg) :					
Uranium	518	464	508	502	245
Plutonium		54	10	16	33
Plutonium composition (%) in new fuel sub assembly (aging time : 2 years)					
Pu^{238}		4.3	5	4.3	4.6
Pu^{239}		48.7	42	36.7	30.3
Pu^{240}		24.9	23	26.7	24.5
Pu^{241}		11.1	10.9	11.0	10.2
Pu^{242}		9.9	18	20.2	29.4
Am^{241}		1.1	1.1	1.1	1.0
Irradiated fuel sub assembly (cooling time : 5 years) plutonium composition (%) :					
Pu^{238}	3.5	5.1	5	4.3	5.8
Pu^{239}	51.0	41.5	42	36.2	11.8
Pu^{240}	24.8	29.0	23	26.5	21.7
Pu^{241}	12.1	12.8	12	12.0	9.6
Pu^{242}	8.6	11.6	18	21.0	51.2

Table 1. (continued.)

	UO_2 open cycle	MOX mono-recycling	MIX Equilibrium	CORAIL 7th Recycle	APA 4th Recycle
Actinide content (%) :					
Pu	1.2	7.9	2.0	2.5	6.7
Np	0.08	0.02	0.07	0.06	0.06
Am	0.07	0.7	0.2	0.2	0.8
Cm	0.01	0.2	0.1	0.05	0.34
Balance (kg/TWh) :					
Plutonium	+26	-53.4	0	0	-70.4
Minor actinides	+3.8	+19	+8.4	+8.8	+16

The most promising concepts were assessed in simplified nuclear fleet scenario studies starting from the current situation up to pseudo steady state. For example, Table 2 gives the annual material flux in a 60-GWe fleet producing 400 TWh.

From Table 2, we can see that:

- The APA concept requires much lower quantities of plutonium fuel to be manufactured than the other concepts (a factor of 15 with respect to CORAIL and a factor of 50 with respect to MIX).
- With respect to the open cycle, the MIX, CORAIL, and APA concepts each reduce the natural uranium and enrichment requirements by about 20%. (A CANDU reactor can reduce these requirements by about 25%.)
- The minor actinide masses produced by the MIX, CORAIL and APA fleets are equivalent but with much more curium in the MIX fleet.
- The MIX, CORAIL and APA fuel cycles reduce the plutonium wastes to almost nothing.

Assuming a nuclear park with 60 GWe producing 400 TWh per year, Figure 5 gives shows the evolution of the plutonium inventory in the cycle (reactors and facilities) for the following PWR scenarios: open cycle, plutonium once through cycling, and plutonium multi-recycling. In 2050, the open cycle has about 630 tons of plutonium, and mono-recycling has about 520 tons. For multiple recycling the plutonium inventory varies between 210 tons (APA and MIX) and 400 tons according to the fuel assembly concept selected.

Table 2. Annual material flux in a 60 GWe fleet producing 400 TWh.

	UO_2 open cycle	MOX mono-recycling	MIX Equilibrium	CORAIL 7th recycle	APA 4th recycle
Burn up (GWd/t)	55	55	55	45	90 (APA) 55 (UO_2)
Reactor ratio (%) PWR(UO_2) PWR(MOX)	 100 -	 58 42	 - 100	 - 100	 70 30
Rod ratio (%) UO_2 MOX	 100 -	 88 12	 - 100	 68 32	 93 7
^{235}U enrichment (%)	4.5	4.5	3.8	4.8	4.5 (UO_2) 3.24 (APA)
Mass fabricated (tons) UO_2 MOX or plutonium in inert matrix	 880 	 770 110	 880	 740 345	 755 19
Natural Uranium (tons) MSWU	8100 6.0	7100 5.3	6800 4.8	7300 5.5	6600 4.8
Mass reprocessed (tons)	-	770	880	1085	775
Wastes (tons) Cooling time: 5 years Pu Np Am Cm Fission products Tc^{99} I^{129} Cs^{135}	 10.6 0.7 0.6 0.1 49 1.2 0.2 0.5	 8.7 0.6 1.3 0.3 49 1.2 0.2 0.6	 0.02 0.6 1.8 0.9 49 1.1 0.2 0.5	 0.03 0.6 2.2 0.5 49 1.3 0.2 0.6	 0.02 0.6 1.7 0.6 47 1.1 0.2 0.6
Plutonium inventory (tons) in the cycle (reactors and facilities), aging time: 2 years, cooling time: 5 years	-	-	220	320*	230

** For the CORAIL fleet 320 tons is a maximum value, corresponding to 45 GWd/t with an initial plutonium content close to the admissible limit. An increase of burnup would require increased ^{235}U enrichment and a lowering of fuel masses stored in the facilities.*

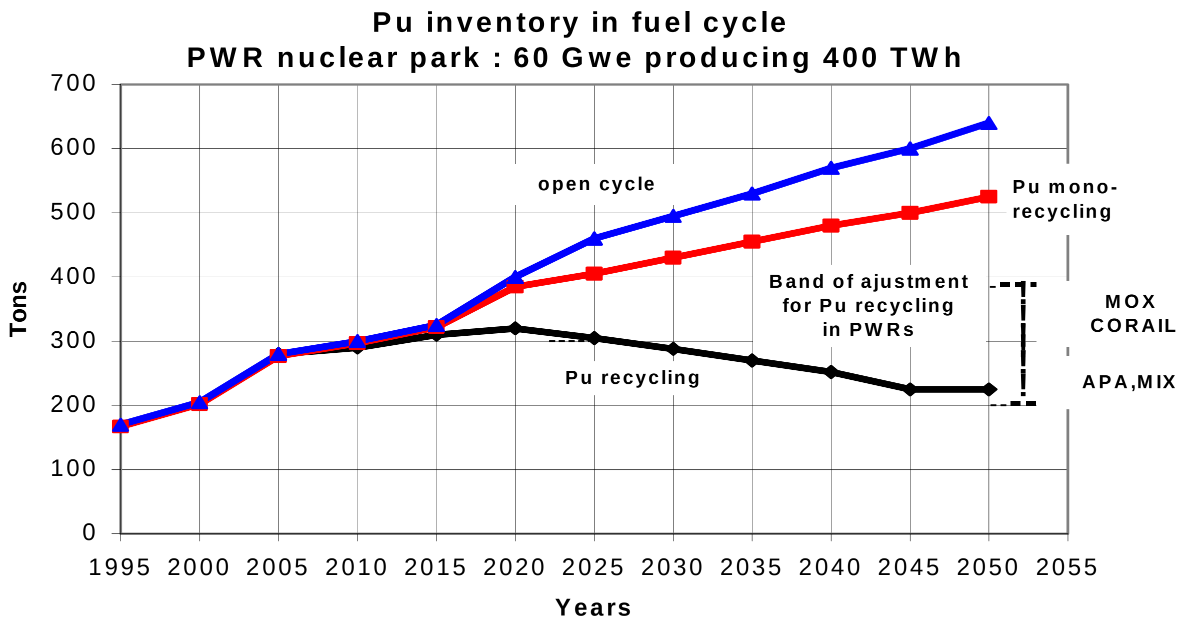

Figure 5. Plutonium inventory versus year.

W10.3 POTENTIAL FOR MEETING GENERATION-IV GOALS

W10.3a. Evaluation Against High Level Criteria

In the following subsections, the ALWRs with Plutonium and Minor Actinide Multi-Recycling concept set is assessed against the Generation IV goals. The advantages and/or disadvantages of this concept set are evaluated relative to a typical Generation III reactor with a once-through uranium fuel cycle. In those areas for which no appreciable differences can be identified between the concept set and the reference, the analyzed concept is rated E (i.e., *Equivalent*) on the score sheet at the end of this appendix. The specific comments under each high-level criterion are related to the Generation IV criteria and metrics by means of a label in parenthesis.

Sustainability–1. Generation IV nuclear energy systems and fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

Plutonium recycle in ALWRs has the following advantage relative to the current LWR once-through uranium fuel cycle with respect to Sustainability-1.

- Advanced fuels for plutonium recycling have the advantage of better using the nuclear resources in recovering the plutonium energy potential rather than managing it like a waste. The savings in natural uranium and SWU due to the use of APA fuels in PWRs are estimated to be 15% - 25% in comparison with the UO_2 open cycle. (SU1-1, SU1-2)

Plutonium recycle in ALWRs has the following relative disadvantages with respect to Sustainability-1:

- No significant disadvantages are noted.

It is concluded that plutonium recycle in ALWRs is somewhat better than the once-through PWR fuel cycle.

Sustainability–2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

Plutonium recycle in ALWRs has the following advantage relative to the current LWR once-through uranium fuel cycle with respect to Sustainability-2.

- A BNFL study found that the use of plutonium recycle in MOX fuel could reduce a PWR's high level waste volume from 40 m^3/GWe-y to 4.9 m^3/GWe-y [Beaumont et al., 1995]. Recent studies also proved the potential for minor actinides incineration (americium and curium). Using APA fuel in 40% of PWR of the park could stabilize the (Pu+Am+Cm) inventory in the cycle. (SU2-1, SU2-2, SU2-3)

Plutonium recycle in ALWRs has the following relative disadvantage with respect to Sustainability-2:

- No significant disadvantage was noted.

Overall, plutonium recycle in ALWRs appears to be significantly better than the current once-through LWR uranium fuel cycle.

Sustainability–3. Generation IV nuclear energy systems and fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

Plutonium recycle in ALWRs has the following advantages relative to the current LWR once-through uranium fuel cycle with respect to Sustainability-3.

- Build-up of actinides (such as ^{238}Pu) with high self-heating and neutron emission reduces the attractiveness of plutonium diversion for weapons use and makes detection of stolen materials easier.

- Incineration of surplus weapons plutonium decreases the stockpile of that fissile material available for diversion. (SU3-1.1, SU3-1.2)

Plutonium recycle in ALWRs has the following relative disadvantage with respect to Sustainability-3.

- The use of reprocessing facilities provides another pathway where diversion could occur. (SU3-2.1)

Overall, it is concluded that plutonium recycle in ALWRs may be somewhat worse in the near term (new pathways for diversion) than the current once-through uranium fuel cycle with respect to Sustainability-3, but it may be somewhat better in the longer term (less total plutonium and very dirty isotopics).

Safety and Reliability–1. Generation IV nuclear energy systems operations will excel in safety and reliability.

Plutonium recycle in ALWRs has the following advantage relative to the current LWR once-through uranium fuel cycle with respect to Safety and Reliability-1.

- No significant advantages relative to current LWRs were noted.

Plutonium recycle in ALWRs has the following relative disadvantages with respect to Safety and Reliability-1.

- The plutonium fraction in the core is limited by safety constraints to about 30% in present PWRs, due to the lower delayed neutron fraction, harder neutron energy spectrum, reduced effectiveness of control rods and boron, and design modifications needed to maintain a negative void coefficient. These issues could affect reactivity control reliability, but they can be accommodated by proper design. (SR2-1.2)

- The buildup of higher actinides during multi-recycling could increase the dose to workers during refueling or during an accident. (SR1-1, SR1-2)

Overall, it is concluded that the plutonium recycling reactors are slightly worse than the present once-through LWRs or ALWRs with regard to Safety and Reliability-1.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

Plutonium recycle in ALWRs has the following advantage relative to the current LWR once-through uranium fuel cycle with respect to Safety and Reliability-2:

- The inert matrix fuel may have lower afterheat than UO_2 fuel. (SR2-1.1)

Plutonium recycle in ALWRs has the following relative disadvantage with respect to Safety and Reliability-2:

- No significant disadvantages were noted.

Overall, it is concluded that the emergency response need for ALWRs with plutonium recycle is comparable to that of ALWRs with a once-through fuel cycle.

Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

Plutonium recycle in ALWRs has the following advantage relative to the current LWR once-through uranium fuel cycle with respect to Safety and Reliability-2:

- No significant advantages were noted.

Plutonium recycle in ALWRs has the following relative disadvantage with respect to Safety and Reliability-2:

- The core would have more TRU present than the once-through fuel cycle core. This would increase the radiotoxicity source term available for potential release during a severe accident, but the increase would probably not change the degree of offsite emergency response required for present LWRs. (SR3-2)

Studies of the offsite consequences of severe accidents are incomplete. With similar fuel forms, cladding, and power density, the release of volatile fission fragments would probably not differ greatly from that in present LWRs or ALWRs. It is concluded that the plutonium recycle concepts are probably equivalent to the current once-through LWR fuel cycle with regard to Safety and Reliability-3.

Economics–1. Generation IV nuclear energy systems will have a clear life cycle cost advantage over other energy sources.

Plutonium recycle in ALWRs has the following advantage relative to the current LWR once-through uranium fuel cycle with respect to Economics-1:

- These fuel cycles result in lower costs for uranium and enrichment. The costs for actinide waste disposal may be reduced if the actinides are multiply recycled and high decontamination factors can be attained during reprocessing. (EC-3)

Plutonium recycle in ALWRs has the following relative disadvantage with respect to Economics-1:

- The reprocessing plant and remote fabrication plant would increase fuel cycle costs. With the APA concept, however, only about ¼ of the total number of rods would contain recycled plutonium, and the rest could be ordinary UO_2 rods. (EC-1)

Overall, we conclude that the life-cycle costs of the plutonium recycle concepts are highly uncertain, but similar to those of the current LWR once-through fuel cycle with respect to Economics-1.

Economics–2. Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

Plutonium recycle in ALWRs has the following advantage relative to the current LWR once-through uranium fuel cycle with respect to Economics-2:

- No significant advantages were noted.

Plutonium recycle in ALWRs has the following relative disadvantage with respect to Economics-2:

- The fuel fabrication and reprocessing would be more expensive than current LWR once-through fuel fabrication, especially if actinides with high self-heating and neutron emission rates were present in the recycle fuel.

Overall, it is concluded that plutonium recycle in ALWRs is worse than the current once-through LWR fuel cycles with respect to Economics-2.

W10.3b. Strengths and Weaknesses

The strengths of plutonium and minor actinide recycle in ALWRs are:

- Enhanced uranium utilization
- Reduced uranium enrichment requirements facilitated by recycle of fissile plutonium

- Reduced waste, especially if actinides are recycled with high decontamination factors
- Possibility to burn surplus weapons plutonium
- The APA concept could produce plutonium with a high content of actinides with self-heating and neutron emission, making that plutonium unattractive for diversion
- Reactor physics codes and data sets are now quite accurate, so the core performance and isotope production and destruction can be predicted with confidence [D'Hondt 2001].

The weaknesses of plutonium recycle in ALWRs are:

- Reprocessing facilities open another pathway for diversion of nuclear materials
- Added fuel-cycle cost of reprocessing facilities
- More expensive fuel fabrication facilities
- Improved decontamination factors for minor actinides would be required for effective multiple recycling.

W10.4 TECHNICAL UNCERTAINTIES

W10.4.a. Research and Development Needs

Fuel assembly manufacturing feasibility has been acquired for oxide-fuel based concepts (MIX and CORAIL), and this type of fuel can be manufactured quite swiftly. The CORAIL assembly design needs to be optimized to limit power peaks between UO_2 and MOX rods. Studies of core transients and of high burnup assemblies (> 45 GWd/t) are also needed.

For the APA assembly, rod manufacture feasibility has not yet been demonstrated. The mechanical and thermohydraulic design also needs verification. Neutronics qualification programs will follow the core physics studies.

In general, these concepts are mostly at early stages and could benefit from additional R&D in fuel assembly design, core transient studies, thermal-hydraulics modeling, fuel fabrication technology, reprocessing technology, effects of high burnup, waste flow analysis, and cost estimates.

W10.4.b. Institutional issues – Licensability and Public Acceptance

The MIX and CORAIL cores are designed to fit within a standard 17x17 PWR core, which should simplify licensing. Possible public concerns about reprocessing and actinide inventory in the core should be offset by the burnup of fissile plutonium and by the reduction of actinide waste streams requiring high-level waste disposal.

W10.4.c. Time-Line for Deployment

It is expected that plutonium and minor actinide multi-recycling could be considered for early deployment (<2015) for MIX and CORAIL, and near term deployment (2025) for APA.

W10.5 INITIAL ASSESSMENT: OBSERVATIONS AND CONCLUSIONS

This range of advanced assembly concepts shows that, from the reactor core physics aspect, solutions for multi-recycling of plutonium in PWRs should be possible. Options range from a concentration of plutonium in a small number of rods (APA, DUPLEX, CORAIL), with or without recourse to an inert matrix, to total dispersion of plutonium throughout the assembly (MIX), with various consequences on manufacturing, plutonium consumption and minor actinide production.

Some of these concepts (MIX) have been subject to detailed studies demonstrating their feasibility. For others (DUPLEX, APA, CORAIL), studies are underway or are awaiting scheduling in order to make a decision concerning their scientific feasibility. All of these solutions require technological validation (manufacturing, behavior under irradiation, etc.) before a decision can be made concerning their technical feasibility. The evaluations are summarized on the attached Top-Tier Screening Sheet for the Advanced Light Water Reactors with Plutonium and Minor Actinide Multi-Recycling concept set.

Multiple recycle of plutonium in ALWRs could enhance nuclear power by appreciably increasing the energy available from uranium resources. It could also reduce the burden of the high level waste to be disposed, especially if minor actinides could be recycled with high decontamination factors. In comparison with the current LWR once-through fuel cycle, the use of reprocessing adds a pathway for possible diversion of fissile materials. The need for reprocessing facilities and for well-shielded fuel fabrication facilities would add to fuel cycle costs. These costs for TRU waste minimization could be partially offset by the reduction of the waste disposal costs, which have not been quantified in the present analysis.

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W10.7 TOP-TIER SCREENING TABLE - ADVANCED LIGHT WATER REACTORS WITH PLUTONIUM AND MINOR ACTINIDE MULTI-RECYCLING

Summary Evaluation: __x__ Retain ____ Reject

Goal		- -	-	+	++	Comments
SU-1	Fuel Utilization					Reduced uranium ore and SWU requirements (15 to 25%) for the APA fuel design.
SU-2	Nuclear Waste					Incineration of minor actinides and plutonium.
SU-3	Proliferation Resistance					Reprocessing and plutonium fuels fabrication activities provide pathways where diversion can occur. However, the buildup of Pu actinides with self-heating and spontaneous neutrons and the reduction in plutonium stockpiles significantly helps the long-term problem.
S&R-1	Worker Safety and Reliability					Lower delayed neutron fraction, reduced control rod worth, less negative void coefficient.
S&R-2	CDF		E			The inert matrix fuel has less decay heat
S&R-3	Mitigation		E			The core has more TRU and an increased source term
E-1	Life-Cycle Cost					Increased reprocessing and fabrication costs vs. savings on waste disposal.
E-2	Capital Cost and Financial Risk					Costs of reprocessing and plutonium fuel fabrication facilities.

Appendix W11

Assessment of the Use of U-Np-Pu Concept

December 2002

CONTENTS

ABSTRACT

The use in light water reactors of a U-Np-Pu fuel containing 2% ^{237}Np has been proposed. The results presented by the developer of this fuel cycle concept show that while high burnups can be achieved (100-200 MWD/kgHM), the necessary uranium enrichments approached 20%. In terms of waste management, the benefits of ^{237}Np burning are modest at best, both because of the additional neptunium produced in the cycle and because of the ^{237}Np produced in the geological repository through the decay of ^{241}Am.

PRODUCTION OF ^{237}Np

In Light Water Reactors, the primary source of ^{237}Np is via a double neutron capture by ^{235}U, as shown in the reactions below. There is also a small contribution through the ^{238}U(n,2n) reaction. In fast reactors, the ^{238}U(n,2n) reaction predominates.

$$^{235}U(n,\gamma)\xrightarrow{98.6\ b}{}^{236}U(n,\gamma)\xrightarrow{5.2\ b}\quad {}^{237}U \quad \xleftarrow{0.013\ b}{}^{238}U(n,2n)$$

$$\beta^- \quad \downarrow \quad 6.75\ d$$

$$^{237}Np$$

In the long term, ^{237}Np is also produced through the decay of ^{241}Am, as shown below.

$$\begin{array}{lcl} & ^{241}Pu & \\ \beta^- & \downarrow & 14.4\,yr \\ & ^{241}Am & \\ \alpha & \downarrow & 432.7\,yr \\ & ^{237}Np & \end{array}$$

Thus the inventory of ^{237}Np in recently discharged fuel is about 750 g/t heavy metal (HM). If the fuel is not reprocessed, the ultimate inventory of ^{237}Np is about 1770 g/ tHM. If the fuel is reprocessed, the amount of ^{237}Np in a geological repository depends on:

- Whether the neptunium is recycled as suggested in this concept
- Whether the plutonium is recycled before the ^{241}Pu can decay to ^{241}Am
- Whether the ^{241}Am is sent to the repository or recycled.

The half-life of ^{237}Np (2.14 million years) is second only to ^{247}Cm among the elements above uranium. Because of its geological mobility and biological activity, ^{237}Np contributes the largest dose from a repository for once-through fuel for times greater than 50,000 years. Thus, any fuel cycle that reduces the amount of ^{237}Np going to the repository will reduce the long-term risk.

However, as shown above, if spent fuel is allowed to decay for 30 or more years before reprocessing, most of the long-term ^{237}Np waste is in the form of ^{241}Am. Reactor concept W15 states that the Am, Cm and fission products are removed, and presumably sent to a geological repository. Thus, any ^{241}Am sent to the repository would decay to ^{237}Np with a 432.7-year half-life. Therefore, after a thousand years a significant inventory of ^{237}Np would be present in the repository, even if the initial ^{237}Np were removed. The 433-year half-life of ^{241}Am is too long to allow decay before reprocessing.

Other Uses of ^{237}Np

Neptunium-237 is the target material for the production of ^{238}Pu, the most widely used isotope in powering radioisotope thermal generators and heaters for deep space probes. Plutonium-238 is produced in the following reaction and decay.

$$
\begin{array}{lcl}
{}^{237}Np\,(n,\gamma) \xrightarrow{169\ b} & {}^{238}Np & \\
\beta^{-} & \downarrow & 2.1\ d \\
 & {}^{238}Pu & \\
\alpha & \downarrow & 89.6\ yr \\
0.5\ W/g & &
\end{array}
$$

Assessment of Concept W15

This reactor/fuel concept aims to use a neptunium-added fuel to:

- Achieve high burnups (100–200 MWd/kgHM)
- Decrease the amount of reprocessing wastes by achieving higher burnups (i.e., less fuel to process)
- Decrease spent fuel radiotoxicity through conversion/transmutation of the neptunium
- Increase the ^{238}Pu content of the fuel through conversion, thus increasing the intrinsic proliferation resistance of the discharged fuel.

It appears that neptunium also demonstrates burnable poison properties, thus reducing the amount of added burnable poisons needed. However, the proposed fuel has increased enrichment requirement for the amount of neptunium to be added to the core, which will increase the fuel cycle costs.

In certain systems, successive neutron captures will convert the neptunium into the fissile plutonium isotopes, and compensate for the depletion of these fissile isotopes. This can reduce the reactivity swing with burnup.

Available ^{237}Np Inventory Compared to Fuel Cycle Needs

Assuming that the total spent fuel inventory in the United States is 70,000 MTU, then the current (estimated) inventory of unseparated ^{237}Np in spent nuclear fuel (in the US) is approximately 50 metric tons. The proposed project aims to use approximately 2% neptunium in the fuel, which could fuel approximately 28 core-loadings of a typical 1000 MW_e reactor using the current U.S. neptunium reserves in spent fuel; or supply 86 one-third core fuel batches for a typical 1000-MW_e reactor. This implies that a certain number of dedicated reactors could be built for neptunium destruction, but will need to use an alternative fuel as the neptunium reserves are depleted. Note however, that even in the proposed reactor and fuel cycle, there is still a neptunium residual at the end of life that will need to be recycled numerous times.

It is known that the stockpiles of ^{237}Np are decreasing rapidly, and the situation is unlikely to change in the absence of further aqueous reprocessing. Any further needs will have to be met through foreign purchase agreements. With this in mind, recent discussions with persons involved in the US ^{238}Pu program have indicated that neptunium is too valuable to use as a reactor fuel. Rather, it should be stockpiled, converted, and used for programs requiring the use of plutonium in radioisotope thermal generator (RTGs) and other thermal energy conversion systems.

Detailed Concerns

Wider Lattice. The cover page explicitly states a certain burnup and the use of a "widened" lattice. This would imply a larger moderator to heavy-metal ratio and a softer spectrum. However, the lattice dimensions and/or the moderator to heavy-metal ratio are never given. Instead, the authors present results of typical light-water-reactor, and harder spectrum cores. Future work should include the performance and reactivity coefficients of specific core designs fueled by U-Np-Pu.

High Burnup. Burnups up to 200 MWd/kgHM are stated, but can also be achieved by using 20% enriched fuel without the addition of neptunium. Neptunium-237 adds no benefit to the reactivity limited burnup. Other burnups are also discussed, but again the enrichment of the fuel is greater than 10%. However, the conversion of the neptunium to Pu-238 *in situ* will add a measure of proliferation resistance.

In-Growth of Neptunium. While the project seems to imply that this could be a way to reduce the amount of neptunium that would be sent to the repository, the authors fail to recognize that a significant amount of neptunium is produced in the decay of ^{241}Pu and ^{241}Am. According to the concept, some of the minor actinides (Am, Cm) and fission products would be removed from irradiated fuel, while the Np, Pu, and U are returned back into the cycle. The separated americium would simply create more neptunium, which does not solve the problem addressed.

CONCLUSIONS

The only perceived benefits of using this particular fuel would be near-term neptunium reduction in the waste. The present U.S. inventory of ^{237}Np would supply only 28 core loadings of a typical 1000-MW_e reactor. In addition, other programs such as space exploration that require the use of neptunium may likely use a significant portion if any is separated from current and future spent fuel. In order to meet other Generation IV goals, a compelling analysis based on the separation costs, enrichment requirements, the neptunium supply, and the value of ^{237}Np for other uses needs to be performed.

Appendix W12

Summary of the Key Features and Initial Assessment of the Gen-IV Water-Cooled Reactor Concepts (Condensed Version)

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Group	Gen-IV Designation	Proposer (Affiliation, Country)	Size / Design Approach	Coolant / Coolant State	Cycle	Thermal Efficiency (*3)	Primary Circuit Layout / Mode of Circulation	Moderator / Moderator State	Spectrum
INTEGRAL PRIMARY-SYSTEM REACTORS	**W18 (IRIS)**	Carelli (Westinghouse, USA)	100-300MWe / Modular	Light Water / Pressurized (15.5MPa)	Indirect	Intermediate (33.3%)	Integrated in single vessel / Forced	Light Water / High Pressure	Thermal
	W10 (SMART)	Chang (KAERI, South Korea)	330MWth / Modular	Light Water / Pressurized (15.0MPa)	Indirect	Intermediate (about 30%)	Integrated in single vessel / Forced	Light Water / High Pressure	Thermal
	W14 (CAREM)	Beatriz-Ramilo (CNEA, Argentina)	100-150MWe / Modular	Light Water / Pressurized	Indirect	Not discussed	Integrated in single vessel / Natural	Light Water / High Pressure	Thermal
	W26 (Multi-Application Small LWR)	Modro (INEEL, USA)	35MWe / Modular	Light Water / Pressurized (10.5MPa)	Indirect	Low (23.2%)	Integrated in single vessel / Natural	Light Water / High Pressure	Thermal
	W16 (PSRD)	Ishida (JAERI, Japan)	100MWth / Modular	Light Water / Pressurized (3.0MPa)	Indirect	Not Discussed	Integrated in single vessel / Natural	Light Water / High Pressure	Thermal
	W17 (MRX - Ship Propulsion)	Ishida (JAERI, Japan)	100MWth / Modular	Light Water / Pressurized (12.0MPa)	Indirect	Not Discussed	Integrated in single vessel / Forced	Light Water / High Pressure	Thermal
	W25 ("Daisy")	Buongiorno (INEEL, USA)	50-150MWe / Modular	Light Water / Boiling (7.4MPa)	Indirect	Low (29.4%)	Integrated in single vessel / Natural	Light Water / High Pressure	Thermal
LOOP PWRs	**W11 (BLOC)**	Park (KAERI, South Korea)	>1500MWe / Modular	Light Water / Pressurized (15.0MPa)	Indirect	Intermediate (same as ALWRs)	Loop / Forced	Light Water / High Pressure	Thermal
	W3 (MARS)	Sorabella (U-Rome, Italy)	150MWe	Light Water / Pressurized (7.5MPa)	Indirect	Low (25%)	Loop / Forced	Light Water / High Pressure	Thermal
	W29 (RAM)	Novelli (POLIMI, Italy)	Not discussed	Light Water / Pressurized (7.5MPa)	Indirect	Not discussed	Loop / Forced	Light Water / High Pressure	Thermal
SIMPLIFIED BWRs	**W7 (SMART)**	Khatib-Rahbar (Energy Research, USA)	50-300MWe / Modular	Light Water / Boiling	Direct	Intermediate (Same as current BWRs)	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2)	Light Water / High Pressure	Thermal
	W8 (SBWR)	Ishii et al. (Purdue Univ., USA)	50MWe / Modular	Light Water / Boiling (7.2MPa)	Direct	Intermediate (30.3%)	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2)	Light Water / High Pressure	Thermal
	W23 (LSBWR)	Heki and Nakamaru (Toshiba, Japan)	300MWe / Simplification and long operating cycles	Light Water / Boiling (7.0MPa)	Direct	Intermediate (Same as current BWRs)	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2) One building for reactor and turbines	Light Water / High Pressure	Thermal
	W22 (Desalination)	Kataoka (Toshiba, Japan)	589MWth / Existing Technologies	Light Water / Boiling (7.0MPa)	Direct	Intermediate (33-35% for electricity) High for desalination	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2)	Light Water / High Pressure	Thermal
	W13 (ESBWR)	Rao (GE, USA)	1380MWe / Monolithic	Light Water / Boiling	Direct	Intermediate (34.5%)	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2)	Light Water / High Pressure	Thermal
PRESSURE-TUBE REACTORS	**W6 (CANDU NG)**	Duffey (AECL, Canada)	600MWe / Monolithic	Light Water / Pressurized (13MPa)	Indirect	High (>35%)	Loop with CANDU-like pressure tubes / Forced	Heavy Water / Low Pressure	Thermal
	W28 (Passive Pressure Tube LWR)	Hejzlar, Todreas, and Driscoll (MIT, USA)	1000MWe / Monolithic	Light Water / Pressurized (15MPa)	Indirect	Intermediate (32%)	Loop with CANDU-like pressure tubes / Forced	Light water and graphite in pressure tubes / High pressure + graphite reflector (and low pressure water ring in wet version)	Thermal
	W5 (Seed and Blanket Pressure Tube LWR)	Kim (U-Kyung Hee, South Korea)	570 MWe	Light Water / Pressurized	Indirect	Intermediate (32%)	Loop with CANDU-like pressure tubes / Forced	Light water and graphite in pressure tubes (dry calandria)/ High Pressure	Thermal
SUPERCRITICAL-WATER REACTORS	**W21 (SCPR)**	Kataoka, Oka, Yoshida, Mimya, & Shiga (Toshiba, etc. Japan)	Size is flexible	Light Water / Supercritical (25MPa)	Direct	Very High (>40%)	Loop (direct cycle) (*1) / Forced (but no re-circulation pumps) (*2)	Light Water / Supercritical Pressure	Thermal (can also be fast)
	TWG1 (Fast Reactor)	Wan (U-Michigan, USA)	1500MWe / Monolithic	Light Water / Supercritical (25MPa)	Direct	Very High (>40%)	Loop (direct cycle) (*1) / Forced	...	Fast
	W6 (Supercritical CANDU)	Corradini (U-Wisconsin, USA) Duffey (AECL, Canada)	400-600MWe / Monolithic	Light Water / Supercritical (25MPa)	Direct	Very High (41% for the Mark 1 and >44% for the ALX2)	Loop with thermally insulated Zircaloy pressure tubes / Forced	Heavy Water / Low Pressure	Thermal
	W2 (MARBLE Fuel)	Tsiklauri (PNNL, USA)	240MWe	Light Water / Supercritical (24MPa)	Direct	Very High (>40%)	Loop (direct cycle) (*1) / Forced	Light Water / Supercritical Pressure	Thermal
HIGH-CONVERSION REACTORS	**W20 (ISPWR/IMR)**	Makihara (Mitsubishi Heavy Industries, Japan)	350MWe / Modular	Heavy Water / Pressurized (15.5MPa)	Indirect	Intermediate (35%)	Integrated in single vessel / Natural	...	Fast
	W19 (SSBWR)	Ohtsuka (Hitachi, Japan)	434MWth / Modular	Heavy Water + Light Water/ Boiling (12.0MPa); Dilution of the heavy water with light water with burnup	Indirect	Intermediate	Integrated in single vessel / Natural	Heavy Water + Light Water / High Pressure	Fast - Epithermal
	TWG 6 (Fast Spectrum)	Diamond (BNL, USA)	4000MWth / Monolithic	Light Water / Boiling (7.0MPa)	Direct	Intermediate (34%)	Loop (direct cycle) (*1) / Forced	...	Fast
	W9 (ABWR-II)	Mochida (Hitachi, Japan)	1500-1700MWe / Monolithic	Light Water / Boiling	Direct	Intermediate (34%)	Loop (direct cycle) (*1) / Forced	...	Fast
	W24 (RMWR)	Iwamura (JAERI, Japan)	1000 MWe	Light Water, Heavy Water/ Boiling, Pressurized	...	...	...	Light Water or Heavy Water/Boiling or Pressurized	Fast
	W30 (RMWR-2)	Okubo (Japan)	1000 MWe	Heavy Water/ Pressurized	Indirect	Intermediate	Loop PWR	Heavy Water/ Pressurized	Fast
	W27 (BARS)	Hiraiwa (Toshiba, Japan)	>1300MWe / Monolithic (same as ABWR)	Light Water / Boiling	Direct	Intermediate (34%)	Loop (direct cycle) (*1) / Forced	...	Fast
PEBBLE FUEL REACTORS	**W1 (MARBLE Fuel)**	Tsiklauri et al. (PNNL, USA)	200MWe	Light Water / Boiling (7.0MPa)	Direct	Intermediate (30%)	Loop (direct cycle) (*1) / Forced	Light Water / High Pressure	Thermal
	W2 (MARBLE Fuel)	Tsiklauri et al. (PNNL, USA)	240MWe	Light Water / Supercritical (24MPa)	Direct	Very High (up to 45%)	Loop (direct cycle) (*1) / Forced	Light Water / High Pressure	Thermal
	W4 (Fluidized Bed)	Sefidvash (UFRGS, Brazil)	1MWe per assembly / Modular	Light Water / Pressurized	Indirect	Intermediate	Integrated in single tube / Forced	Light Water / High Pressure; Graphite or Water Reflector	Thermal
ADVANCED LIGHT WATER REACTORS WITH THORIUM-URANIUM FUEL	**TWG 7 (Homogeneous Thorium Fuel Cycles)**	MacDonald (INEEL, USA)	...	Light Water / Pressurized, Boiling	...	...	...	Light Water / High Pressure	Thermal
	TWG 8 (Seed and Blanket Thorium Fuel Cycles)	Diamond (BNL, USA)	...	Light Water / Pressurized, Boiling	...	...	...	Light Water / High Pressure	Thermal
	TWG 5 (Shippingport / Thermal Breeder)	MacDonald (INEEL, USA)	...	Light Water / Pressurized	Indirect	Intermediate	Loop PWR	Light Water / High Pressure	Thermal
ADVANCED WATER-COOLED REACTORS WITH DRY RECYCLE OF SPENT LWR FUEL	**W12 (DUPIC)**	Yang (KAERI, South Korea)	...	Light Water, Heavy Water	...	...	...	Light Water, Heavy Water	Thermal
ADVANCED LIGHT WATER REACTORS WITH PLUTONIUM AND ACTINIDE MULTI-RECYCLE	**TWG 2 (MIX)**	Vasile (CEA, France)	...	Light Water / Pressurized	Indirect	Intermediate	Loop PWR	Light Water / High Pressure	Thermal
	TWG 3 (CORAIL)	Vasile (CEA, France)	...	Light Water / Pressurized	Indirect	Intermediate	Loop PWR	Light Water / High Pressure	Thermal
	TWG 4 (APA)	Vasile (CEA, France)	...	Light Water / Pressurized	Indirect	Intermediate	Loop PWR	Light Water / High Pressure	Thermal
	W15 (U-Pu-Np FUEL CYCLE)	Saito (Tokyo Institute of Tech, Japan)	...	Light Water, Heavy Water	...	...	...	Light Water, Heavy Water	Thermal, Fast

NOTES

(*1) In a direct-cycle reactor the primary coolant circulates in an out-of-vessel loop and thus has to be considered by definition a loop-type reactor.
(*2) Although the circulation of the primary coolant within the vessel is not pump-driven, the primary coolant in the external loop is pumped.
(*3) The following definitions are adopted for the thermal efficiency: Low (<30%), Intermediate (30-35%), High (35-40%), Very High (>40%)
(*4) The following definitions are adopted for the irradiation cycle length: Short (<1yr), Intermediate (1-2yrs), Long (>2yrs).
(*5) In this field "eliminated" means the possibility of a certain accident is eliminated by design; "mitigated" means the consequences of a certain accidents are mitigated by passive means.
(*6) The following definitions are adopted for the level of R&D required to develop the concept: Minimal = fuel and materials are well-established; Modest = fuel and/or materials need development and testing; Extensive = fuel and/or materials need significant, long-term development and testing.

Appendix 4E2.B SUMMARY OF THE KEY FEATURES AND INITIAL ASSESSMENT OF THE GEN IV WATER-COOLED REACTOR CONCEPTS

Preferred Fuel (Status)	Backup Fuel (Status)	Irradiation Cycle / Refueling (1)	Cladding Materials (Status)	Reactivity Control	Decay Heat Removal System	Containment	Improved Safety Characteristics (1)	Proliferation Characteristics	Resource Utilization	Economics Characteristics
LEU-UO2 Rods (Mature)	MOX Rods (Mature)	Long / offline	Zircaloy (Mature)	Top-entry CRs, Burnable poisons, No boron	[illegible]	[illegible]	LOCAs (large and small) eliminated, LOFAs mitigated	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Minimal R&D costs, nuclear island simplification, factory fabricability
[illegible] LEU Rods (Mature)	...	Long / offline	Zircaloy (Mature)	Top-entry CRs, Burnable poisons, No boron	Passive (emergency heat exchangers)	[illegible]	Large LOCAs eliminated, LOFAs mitigated, [illegible]	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Minimal R&D costs, nuclear island simplification, factory fabricability
LEU-UO2 Rods (Mature)	...	Intermediate / offline	Zircaloy (Mature)	Top-entry CRs, Burnable poisons, Passive scram, No boron	Passive (HP emergency condensers)	Small (with suppression pool)	Large LOCAs, LOFAs, CRE eliminated	Comparable to current LWRs	Comparable to current LWRs	Low power per module, minimal R&D costs, nuclear island simplification, factory fabricability
LEU-UO2 Rods (Mature)	LEU-ThO2 Rods (Development Required)	Long / offline	Zircaloy (Mature)	Not discussed	[illegible]	[illegible]	Large LOCAs, LOFAs eliminated	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	[illegible]
[illegible]	...	Long / offline	Zircaloy (Mature)	Top-entry CRs, No boron	[illegible]	[illegible]	[illegible]	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	[illegible]
[illegible]	...	Long / offline	Zircaloy (Mature)	Top-entry CRs	[illegible]	[illegible]	[illegible]	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	[illegible]
[illegible]	MOX Rods (Mature)	Long / offline	Zircaloy (Mature)	Bottom-entry CRs, Burnable poisons	[illegible]	[illegible]	[illegible]	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	[illegible]
[illegible]	...	Long / offline	[illegible]	Top-entry CRs, No boron	[illegible]	[illegible]	[illegible]	Low Pu production + dirty Pu isotopics	Th cycle	[illegible]
LEU-UO2 Rods (Mature)	...	[illegible]	Zircaloy (Mature)	[illegible]	[illegible]	[illegible]	[illegible]	Comparable to current LWRs	Comparable to current LWRs	[illegible]
LEU-UO2 Rods (Mature)	...	Not discussed	Zircaloy (Mature)	[illegible]	Not discussed	Not Discussed	Passive shutdown	Comparable to current LWRs	Comparable to current LWRs	Elimination of the control system
LEU-UO2 rods (Mature)	LEU-ThO2 Rods (Development Required)	Long / offline	Zircaloy (Mature)	Top-entry CRs	[illegible]	[illegible]	[illegible]	Comparable to current LWRs + long irradiation cycle	[illegible]	[illegible]
[illegible]	LEU Rods (Mature)	Long / offline	Zircaloy (Mature)	Bottom-entry CRs	[illegible]	Small	[illegible]	Low Pu production + dirty Pu isotopics	Th cycle	[illegible]
LEU-UO2 rods (Mature)	[illegible]	[illegible]	[illegible]	Bottom-entry CRs, Burnable poisons	[illegible]	[illegible]	[illegible]	[illegible]	Comparable to current LWRs	[illegible]
LEU-UO2 rods (Mature)	...	Long / offline	Zircaloy (Mature)	Bottom-entry CRs	[illegible]	[illegible]	[illegible]	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	[illegible]
LEU-UO2 rods (Mature)	MOX Rods (Mature)	Intermediate / offline	Zircaloy (Mature)	Bottom-entry CRs	[illegible]	[illegible]	[illegible]	Comparable to current LWRs	Comparable to current LWRs	Minimal R&D costs
UO2-ThO2 Fuel Rods (Development Required)	LEU-UO2 Rods (Mature)	Intermediate / continuous online refueling	Coated Zr-2.5%Nb for Mark 1, not discussed for ALX2 (Development of high temperature cladding required)	Online refueling	Active (traditional CANDU ECCS)	Yes	CHF eliminated, small water inventory and no natural circulation paths	Low Pu production + dirty Pu isotopics	Th cycle (neutron efficient because of moderation)	Very high thermal efficiencies, smaller BOP components and no steam generators or separators, possible material problems, thicker pressure tubes and piping
For dry version, TRISO Coated UO2 in a compact in a block of graphite coated with SiC (development required)	For wet version, CANDU type fuel bundle with central rings replaced with a SiC coated graphite plug (development required)	Intermediate / continuous online refueling	SiC for dry version, ZrC or SiC for wet version (Major development needed)	Scram by flooding the calandria tank, Online refueling	Passive (flooding of the calandria tank for dry version, natural circulation of the water in the annular ring for the wet version)	Yes (Cooled by natural circulation of air on the outside)	Reflooding of the primary system unnecessary, tritium eliminated	Comparable to current LWRs	High burnup and possible reuse of the fuel in CANDUs	Heavy water replaced with light water, boron is eliminated, safety systems are reduced
Seed: uranium-15%zirconium fuel rods, Blanket: BISO coated Th02 and UCO fuel particles in graphite matrix compacted into pellets (development required)	MOX, DUPIC	Long / continuous online refueling	Zircaloy (Mature)	Scram by flooding the calandria tank, Online refueling	Passive (flooding of the calandria tank)	Yes (cooled by natural circulation of air)	Reflooding of the primary system unnecessary, tritium eliminated	Low Pu production + dirty Pu isotopics	Th cycle + can burn Pu	Heavy water replaced with light water, boron is eliminated, safety systems are reduced
UO2 Rods (Mature)	MOX Rods (Mature)	Intermediate (1-2)yrs / Offline	Stainless steel, high nickel alloys, or titanium alloys (Major Development Required)	Top-entry CRs, Burnable poisons	Active (but passive systems can also be used)	Small (with suppression pool)	CHF eliminated, negative void coefficient, but small water inventory and no natural circulation paths	Comparable to current LWRs	High conversion/breeding is achievable more easily than in current LWRs	Very high thermal efficiencies, smaller BOP components and no steam generators or separators, possible material problems, thicker reactor pressure vessel and piping, but less total material
U- and Th-based nitride or metal (Major Development Required)	MOX Rods (Mature)	Long / offline	High nickel alloys or austenitic-martinsitic stainless steels (Major Development Required)	Top-entry CRs, Feed water flow	Active (similar to ABWR)	Small	CHF eliminated, small water inventory and no natural circulation paths	No Pu production (with a breeding ratio of 1) + dirty Pu isotopics	Can burn actinides from spent LWR fuel (no need to mine uranium)	Very high thermal efficiencies, no steam generators or separators, complicated ECCS, possible material problems, thicker reactor pressure vessel and piping
UO2-ThO2 Fuel Rods (Development Required)	LEU-UO2 Rods (Mature)	Intermediate / continuous online refueling	Coated Zr-2.5%Nb for Mark 1, not discussed for ALX2 (Development of high temperature cladding required)	Online refueling	Active (traditional CANDU ECCS)	Yes	CHF eliminated, small water inventory and no natural circulation paths	Low Pu production + dirty Pu isotopics	Th cycle (neutron efficient because of moderation)	Very high thermal efficiencies, smaller BOP components and no steam generators or separators, possible material problems, thicker pressure tubes and piping
Fluidized Bed SiC-PyC-Coated UO2 Particles (Major Development Required)	...	Intermediate / continuous online refueling	SiC and pyrocarbons (Major Development Required)	Bottom-entry CRs, Online refueling	Passive (Radial Conduction)	Not Discussed	CHF eliminated + good FP retention in the fuel + decay heat removed by radiation and conduction	Coated particles hard to reprocess	Comparable to current LWRs	Very high thermal efficiencies, smaller BOP components and no steam generators or separators, possible material problems, thicker reactor pressure vessel and piping, but less total material
UO2 Rods (Mature)	MOX Rods (Mature)	Long / offline	Not Discussed	Top-entry CRs, Low control requirements from reactivity swing minimization	Passive (HP emergency condensers)	Small	Large LOCAs, LOFAs eliminated	Comparable to current FBRs	High conversion U/Pu cycle	Cost of heavy water coolant, heavy water loss
UO2 rods (Mature)	...	Long / offline	Metal	Top-entry CRs, Low control requirements from reactivity swing minimization	Passive (depressurization + in-containment heat pipes)	Small (with suppression pool)	Large LOCAs, LOFAs eliminated	Comparable to current LWRs + long irradiation cycle	High conversion U/Pu cycle + can burn MA and FPs	Heavy water dilution, relatively high fuel costs, relatively low power, simplified ECCS, reduced number of CRDs, and high capacity factors
ThO2-UO2-PuO2 Rods (Development Required)	Nitride Rods (Major Development Required)	Long / offline	Not Discussed	Streaming channels for negative void coefficient, Low control requirements from reactivity swing minimization	Not Discussed	Not Discussed	Reactivity swing minimized	Low Pu production + dirty Pu isotopics	Th cycle + can burn Pu	Capital cost similar to ABWR
MOX rods with dry reprocessing	...	Intermediate / offline	Zircaloy (needs to verify performance in fast spectrum)	Bottom-entry CRs with moderator-displacing follower, Re-circulation pumps, Low control requirements from reactivity swing minimization	Active (diesel-driven ECCS) + Passive (in-containment heat exchangers)	Yes	Fast reactor with negative void response + safety similar to ABWR	Dry reprocessing; no actinides separation	High conversion U/Pu cycle + can burn MA and FPs	Capital cost similar to ABWR
MOX fuel in tight low-moderation core	...	Intermediate for breeding, Long for internal conversion / offline	Zircaloy or advanced stainless steel	Low control requirements from reactivity swing minimization, Short fuel rod for negative void coefficient	...	...	Reactivity swing minimization	Comparable to current FBRs	Can breed Pu or high internal-conversion U/Pu cycle	Requires reprocessing
MOX fuel in tight low-moderation core with seed and DU blanket regions	...	Long / offline	Stainless steel	Low control requirements from reactivity swing minimization, internal blanket regions for negative void coefficient	...	...	Reactivity swing minimization	Comparable to current FBRs	Can breed Pu or high internal-conversion U/Pu cycle	Requires reprocessing
MOX rods with dry reprocessing	UO2 rods (Mature)	Intermediate / offline	Not Discussed	Streaming channels for negative void coefficient, Low control requirements from reactivity swing minimization	Similar to ABWR	Similar to ABWR	Fast reactor with negative void response + safety similar to ABWR	Dry reprocessing; no actinide separation	High conversion U/Pu cycle + can burn MA and FPs	Capital cost similar to ABWR
Fluidized Bed of SiC-PyC-Coated UO2 Particles (Fabrication processes need development)	...	Intermediate / continuous online refueling	SiC-PyC in Water (Development Required)	Bottom-entry CRs, Online refueling	Passive (Radial Conduction)	Not Discussed	Good FP retention in the particle fuel	Coated particles hard to reprocess	Comparable to current LWRs	Somewhat higher plant capacity factors with online refueling
Fluidized Bed of SiC-PyC-Coated UO2 Particles (Fabrication processes need development)	...	Intermediate / continuous online refueling	SiC-PyC in Water (Development Required)	Bottom-entry CRs, Online refueling	Passive (Radial Conduction)	Not Discussed	CHF eliminated + good FP retention in the fuel + decay heat removed by radiation and conduction	Coated particles hard to reprocess	Comparable to current LWRs	Very high plant efficiency, somewhat higher plant capacity factors with online refueling
Fluidized Bed of LEU-UO2 Particles (Major Development Required)	...	Not Discussed / continuous on line refueling	Zircaloy (Mature)	Primary coolant flow, Passive scram, No CRs, No boron	Passive (Convection to air or water)	Underground containment	The suspended bed is critical only under certain conditions. Upset conditions lead to a loss of criticality.	Comparable to current LWRs	Comparable to current LWRs	Need a SG, a pump and refueling machine per fuel assembly
Homogeneous U02-Th02 fuel rods (Development Required)	Micro heterogeneous U02-Th02 fuel rods (Major Development Required)	Long / offline	Zircaloy (Mature)	Low control requirements from reactivity swing minimization	...	...	Reactivity swing minimization, significant power peaking when micro-hetrogeneous fuel is used	Low Pu production + dirty Pu isotopics	Th cycle + can burn Pu	High SWU costs
Single bundle with a U-Zr metal fuel seed region and U02-Th02 blanket rods (Major development required for the seed)	Alternate metal fuel seed and thoria-urania blanket assemblies	Long / offline	Zircaloy (Mature)	Low control requirements from reactivity swing minimization	...	...	Reactivity swing minimization, significant power peaking a beginning-of-cycle	Very low Pu production + dirty Pu isotopics	Th cycle + can burn Pu	High SWU costs, uncertain metal fuel fabrication costs
Movable U02 seed region + U02-Th02 blanket region (Development required)	...	Long / offline	Zircaloy (Mature)	No control rods, reactivity controlled by movable seed rods	...	...	Reactivity swing minimization, significant power peaking a beginning-of-cycle	Very low Pu production + dirty Pu isotopics	Th cycle, thermal breeder	Complicated and possibly costly core design
Actinides from LWR spent fuel, dry-reprocessed, resintered in MOX pellets and (if re-enriched) used in LWRs, (if not re-enriched) used in CANDU reactors	...	...	Zircaloy (Mature)	Not discussed	...	...	...	Dry reprocessing; no separation of the actinides from the uranium and most of the fission products	Allows recycle of LWR spent fuel back into a LWR with only a slight addition of fissile material or directly into a CANDU, therefore, there is little or no mining	Disposal of the gaseous and volatile fission products and remote fuel fabrication in a hot cell are expensive
Homogeneous MOX rods with varying Pu content	...	Short / Offline	Zircaloy (Mature)	Pu hardens the spectrum and reduces the worth of the control rods. More control rods required	...	...	Smaller delayed neutron fraction; total load of Pu must be limited	Comparable to current FBRs	Pu recycling	Requires reprocessing
MOX rods + enriched UO2 rods	...	Short / Offline	Zircaloy (Mature)	Pu hardens the spectrum and reduces the worth of the control rods. More control rods required	...	...	Smaller delayed neutron fraction; total load of Pu must be limited	Comparable to current FBRs	Pu recycling	Requires reprocessing
Pu inert matrix annular rods for increased moderation and coolability (Development required) + traditional UO2 rods	...	Short / Offline	Zircaloy (Mature)	Pu hardens the spectrum and reduces the worth of the control rods. More control rods required	...	...	Smaller delayed neutron fraction; total load of Pu must be limited	Comparable to current FBRs	Pu recycling	Requires reprocessing
Uses Np-enriched MOX fuel to extend fuel lifetime and minimize reactivity swing	...	Long / offline	Not Discussed	Low control requirements from reactivity swing minimization	...	...	Reactivity swing minimization	Dirty Pu Isotopics	High BU + recycling of Pu and Np	Requires reprocessing

R&D Needs by reactor concept (*1)	R&D Needs by group	Estimated Time of Deployment	Group Evaluator
Minimal Minimal Minimal Minimal (with UO2 fuel) Modest (with Th fuel) Minimal Minimal Minimal	1) Design and accessibility of in-vessel control rod drives. 2) Design and accessibility of the in-vessel HXs. 3) Long-cycle corrosion control of the cladding materials.	<2015	Carelli (Westinghouse), MacDonald (INEEL), Delmastro (CNEA)
Modest: Th-U metal fuel development, cladding development, passive systems for a large PWR Minimal Minimal	1) Accessibility of the underwater primary-system components. 2) Seismic response of the large distributed loop.	<2015 (with U fuel) >2015 (with Th fuel)	Park (KAERI), Lee (DHICO), Lauret (Framatome)
Minimal (with UO2 fuel) Modest (with Th fuel) Modest: Th fuel development Modest: development of oxide fuel for use well beyond current LWR burnups Minimal Minimal	1) Overall R&D needs are minimal because these are mostly small BWRs with little conceptual innovation. 2) Long-cycle corrosion control of the cladding materials.	<2015 (with U fuel) >2015 (with Th fuel)	Devine (Polestar), Schultz (INEEL), Diamond (BNL)
Modest: development of high pressure and temperature pressure tubes and modified shutdown system components Extensive fuel development Extensive fuel development	1) Development of higher pressure and temperature pressure tubes. 2) Development of metal and TRISO and BISO coated fuels for LWRs.	CANDU-NG <2015, W5 and W28 >2015	Hedges (AECL), Park (KAERI), Lee (DHICO)
Extensive: cladding and internal structural materials development Extensive: cladding and internal structural materials development, metal and nitride fuels development Extensive: cladding materials development, development of very high pressure and temperature pressure tubes Extensive: pebble fuel and structural materials development	1) Development of fuel, cladding and structural materials. 2) Demonstration of adequate reactor performance and safety. 3) Metal and nitride fuels development (for TWG1).	>2015	Was (U-Michigan), Corradini (U-Wisconsin), Smith (Dominion)
Modest: cladding and internal structure materials, nuclear data, controllability (small beta), negative void coefficients, heavy water decomposition Modest: cladding and internal structural materials, nuclear data, negative void coefficients, coolant density wave instabilities, heavy water decomposition Modest: cladding materials, nuclear data, controllability (small beta), negative void coefficients, coolant density wave instabilities, CHF margin and accident coolability in tight cores Modest: cladding materials, nuclear data, controllability (small beta), negative void coefficients, coolant density wave instabilities, CHF margin and accident coolability in tight cores Modest: development of cladding materials for water cooled fast reactors and confirmation of tight lattice core coolability during accidents Modest: development of cladding materials for water cooled fast reactors and confirmation of tight lattice core coolability during accidents Modest: cladding materials, nuclear data, controllability (small beta), negative void coefficients, coolant density wave instabilities, CHF margin and accident coolability in tight cores	1) Demonstration of the reliability of clad and structural materials in a water-cooled fast reactor. 2) Prevention of CHF or overheating in tight-lattice water-cooled cores. 3) Improved nuclear data	>2015	Diamond (BNL), Vasile (CEA)
Extensive: pebble fuel reliability in fully-fluidized water bed, safety testing Extensive: pebble fuel and structural materials development, safety testing Extensive: pebble fuel reliability in fully-fluidized water bed, fuel fabrication technology	1) Demonstration of the reliability of pebble fuel in fully-fluidized beds. 2) Fabrication of large particles by CVD. 3) Fabrication of Zircaloy clad UO2 spherical fuel	>2015	MacDonald (INEEL)
Modest: demonstration testing of thoria-urania fuel, Extensive ; safety testing of micro-heterogeneous fuel Extensive: metal driver fuel development and safety testing Modest: already successful demonstrated	1) Demonstration of homogeneous U-Th oxide fuel. 2) Development of U-Zr metallic fuel for water-cooled reactors. 3) Demonstration of thermal-hydraulic safety in the high-power-peaking heterogeneous fuel.	>2015	MacDonald (INEEL), Diamond (BNL)
Extensive (the process has never been shown to be feasible on a production scale)	1) Handling of waste streams from dry reprocessing. 2) Demonstration of remote oxide-fuel pellet refabrication in hot cells.	<2015	Hedges (AECL), Park (KAERI)
Minimal: MOX fuel is established Minimal: MOX fuel is established Extensive: annular rods with Pu and CeO2 need to be developed Modest: development of Np-enriched MOX fuel that will achieve 200MWd/kg burnup	1) Demonstration of high-Pu core controllability. 2) Development of advanced reprocessing techniques for low-cost, proliferation-resistant multiple recycling of Pu and MA. 3) Development of annular rods for the APA concept.	<2015	Vasile (CEA), Diamond (BNL), MacDonald (INEEL)

Appendix W13

Summary of the Key Features and Initial Assessment of the Gen-IV Water-Cooled Reactor Concepts

December 2002

Appendix W13: SUMMARY OF THE KEY FEATURES AND INITIAL ASSESSMENT OF THE GEN-IV WATER-COOLED REACTOR CONCEPTS

Group	Gen-IV Designation	Proposer (Affiliation, Country)	Size / Design Approach	Coolant / Coolant State	Cycle
INTEGRAL PRIMARY-SYSTEM REACTORS	**W18 (IRIS)**	Carelli (Westinghouse, USA)	100-300MWe / Modular	Light Water / Pressurized (15.5MPa)	Indirect
	W10 (SMART)	Chang (KAERI, South Korea)	330MWth / Modular	Light Water / Pressurized (15.0MPa)	Indirect
	W14 (CAREM)	Beatriz-Ramilo (CNEA, Argentina)	100-150MWe / Modular	Light Water / Pressurized	Indirect
	W26 (Multi-Application Small LWR)	Modro (INEEL, USA)	35MWe / Modular	Light Water / Pressurized (10.5MPa)	Indirect
	W16 (PSRD)	Ishida (JAERI, Japan)	100MWth / Modular	Light Water / Pressurized (3.0MPa)	Indirect
	W17 (MRX - Ship Propulsion)	Ishida (JAERI, Japan)	100MWth / Modular	Light Water / Pressurized (12.0MPa)	Indirect
	W25 ("Daisy")	Buongiorno (INEEL, USA)	50-150MWe / Modular	Light Water / Boiling (7.4MPa)	Indirect
LOOP PWRs	**W11 (BLOC)**	Park (KAERI, South Korea)	>1500MWe / Modular	Light Water / Pressurized (15.0MPa)	Indirect
	W3 (MARS)	Sorabella (U-Rome, Italy)	150MWe	Light Water / Pressurized (7.5MPa)	Indirect
	W29 (RAM)	Novelli (POLIMI, Italy)	Not discussed	Light Water / Pressurized (7.5MPa)	Indirect

SIMPLIFIED BWRs	**W7 (SMART)**	Khatib-Rahbar (Energy Research, USA)	50-300MWe / Modular	Light Water / Boiling	Direct
	W8 (SBWR)	Ishii et al. (Purdue Univ., USA)	50MWe / Modular	Light Water / Boiling (7.2MPa)	Direct
	W23 (LSBWRI)	Heki and Nakamaru (Toshiba, Japan)	300MWe / Simplification and long operating cycles	Light Water / Boiling (7.0MPa)	Direct
	W22 (Desalination)	Kataoka (Toshiba, Japan)	589MWth / Existing Technologies	Light Water / Boiling (7.0MPa)	Direct
	W13 (ESBWR)	Rao (GE, USA)	1380MWe / Monolithic	Light Water / Boiling	Direct

PRESSURE-TUBE REACTORS	**W6 (CANDU NG)**	Duffey (AECL, Canada)	600MWe / Monolithic	Light Water / Pressurized (13MPa)	Indirect
	W28 (Passive Pressure Tube LWR)	Hejzlar, Todreas, and Driscoll (MIT, USA)	1000MWe / Monolithic	Light Water / Pressurized (15MPa)	Indirect
	W5 (Seed and Blanket Pressure Tube LWR)	Kim (U-Kyung Hee, South Korea)	670 MWe	Light Water / Pressurized	Indirect

SUPERCRITICAL-WATER REACTORS	**W21 (SCPR)**	Kataoka, Oka, Yoshida, Moriya, & Shiga (Toshiba, etc. Japan)	Size is flexible	Light Water / Supercritical (25MPa)	Direct
	TWG1 (Fast Reactor)	Was (U-Michigan, USA)	1500MWe / Monolithic	Light Water / Supercritical (25MPa)	Direct
	W6 (Supercritical CANDU)	Corradini (U-Wisconsin, USA), Duffey (AECL, Canada)	400-600MWe / Monolithic	Light Water / Supercritical (25MPa)	Direct
	W2 (MARBLE Fuel)	Tsiklauri (PNNL, USA)	240MWe	Light Water / Supercritical (24MPa)	Direct

HIGH-CONVERSION REACTORS	**W20 (ISPWR/IMR)**	Makihara (Mitsubishi Heavy Industries, Japan)	350MWe / Modular	Heavy Water / Pressurized (15.5MPa)	Indirect
	W19 (SSBWR)	Ohtsuka (Hitachi, Japan)	434MWth / Modular	Heavy Water + Light Water/ Boiling (12.0MPa); Dilution of the heavy water with light water with burnup	Indirect
	TWG 6 (Fast Spectrum)	Diamond (BNL, USA)	4000MWth / Monolithic	Light Water / Boiling (7.0MPa)	Direct
	W9 (ABWR-II)	Mochida (Hitachi, Japan)	1500-1700MWe / Monolithic	Light Water / Boiling	Direct

	W24 (RMWR)	Iwamura (JAERI, Japan)	1000 MWe	Light Water, Heavy Water/ Boiling, Pressurized	…
	W30 (RMWR-2)	Okubo (Japan)	1000 MWe	Heavy Water/ Pressurized	Indirect
	W27 (BARS)	Hiraiwa (Toshiba, Japan)	>1300MWe / Monolithic (same as ABWR)	Light Water / Boiling	Direct

PEBBLE FUEL REACTORS	**W1 (MARBLE Fuel)**	Tsiklauri et al. (PNNL, USA)	200MWe	Light Water / Boiling (7.0MPa)	Direct
	W2 (MARBLE Fuel)	Tsiklauri et al. (PNNL, USA)	240MWe	Light Water / Supercritical (24MPa)	Direct
	W4 (Fluidized Bed)	Sefidvash (UFRGS, Brazil)	1MWe per assembly / Modular	Light Water / Pressurized	Indirect

ADVANCED LIGHT WATER REACTORS WITH THORIUM/ URANIUM FUEL	**TWG 7 (Homogeneous Thorium Fuel Cycles)**	MacDonald (INEEL, USA)	…	Light Water / Pressurized, Boiling	…
	TWG 8 (Seed and Blanket Thorium Fuel Cycles)	Diamond (BNL, USA)	…	Light Water / Pressurized, Boiling	…

	TWG 5 (Shippingport / Thermal Breeder)	MacDonald (INEEL, USA)	…	Light Water / Pressurized	Indirect

ADVANCED WATER-COOLED REACTORS WITH DRY RECYCLE OF SPENT LWR FUEL	**W12 (DUPIC)**	Yang (KAERI, South Korea)	…	Light Water, Heavy Water	…

ADVANCED LIGHT WATER REACTORS WITH PLUTONIUM AND ACTINIDE MULTI-RECYCLE	**TWG 2 (MIX)**	Vasile (CEA, France)	…	Light Water / Pressurized	Indirect
	TWG 3 (CORAIL)	Vasile (CEA, France)	…	Light Water / Pressurized	Indirect
	TWG 4 (APA)	Vasile (CEA, France)	…	Light Water / Pressurized	Indirect
	W15 (U-Pu-Np FUEL CYCLE)	Saito (Tokyo Institute of Tech, Japan)	…	Light Water, Heavy Water	…

NOTES

(*1) In a direct-cycle reactor the primary coolant circulates in an out-of-vessel loop and thus has to be considered by defi

(*2) Although the circulation of the primary coolant within the vessel is not pump-driven, the primary coolant in the extern

(*3) The following definitions are adopted for the thermal efficiency: Low (<30%), Intermediate (30-35%), High (35-40%),

(*4) The following definitions are adopted for the irradiation cycle length: Short (<1yr), Intermediate (1-3yrs), Long (>3yrs

(*5) In this field "eliminated" means the possibility of a certain accident is eliminated by design; "mitigated" means the co

(*6) The following definitions are adopted for the level of R&D required to develop the concept: Minimal = fuel and materi

Thermal Efficiency (*3)	Primary Circuit Layout / Mode of Circulation	Moderator / Moderator State	Spectrum	Preferred Fuel (Status)	Backup Fuel (Status)
Intermediate (33.3%)	Integrated in single vessel / Forced	Light Water / High Pressure	Thermal	LEU-UO2 Rods (Mature)	MOX Rods (Mature)
Intermediate (about 30%)	Integrated in single vessel / Forced	Light Water / High Pressure	Thermal	5%-enriched UO2 Rods (Mature)	…
Not discussed	Integrated in single vessel / Natural	Light Water / High Pressure	Thermal	LEU-UO2 Rods (Mature)	…
Low (23.3%)	Integrated in single vessel / Natural	Light Water / High Pressure	Thermal	LEU-UO2 Rods (Mature)	UO2-ThO2 Rods (Development Required)
Not Discussed	Integrated in single vessel / Natural	Light Water / High Pressure	Thermal	<5%-enriched UO2 Rods (Mature)	…
Not Discussed	Integrated in single vessel / Forced	Light Water / High Pressure	Thermal	4.3%-enriched UO2 Rods (Mature)	…
Low (29.4%)	Integrated in single vessel / Natural	Light Water / High Pressure	Thermal	3%-enriched UO2 Rods (Mature)	MOX Rods (Mature)
Intermediate (same as ALWRs)	Loop / Forced	Light Water / High Pressure	Thermal	ThO2-UO2 dispersed in Zr matrix (Development Required)	…
Low (25%)	Loop / Forced	Light Water / High Pressure	Thermal	LEU-UO2 Rods (Mature)	…
Not discussed	Loop / Forced	Light Water / High Pressure	Thermal	LEU-UO2 Rods (Mature)	…

Intermediate (Same as current BWRs)	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2)	Light Water / High Pressure	Thermal	LEU-UO2 rods (Mature)	UO2-ThO2 Rods (Development Required)
Intermediate (30.3%)	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2)	Light Water / High Pressure	Thermal	5%-enriched ThO2-UO2 Rods (Development Required)	UO2 Rods (Mature)
Intermediate (Same as current BWRs)	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2) One building for reactor and turbines	Light Water / High Pressure	Thermal	LEU-UO2 rods (Mature)	MEU-UO2 rods for very high burnup (Development required)
Intermediate (33-35% for electricity) High for desalination	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2)	Light Water / High Pressure	Thermal	LEU-UO2 rods (Mature)	…
Intermediate (34.5%)	Loop (direct cycle) (*1) / Forced feed water (but natural circulation in the core and no re-circulation pumps) (*2)	Light Water / High Pressure	Thermal	LEU-UO2 rods (Mature)	MOX Rods (Mature)

High (>36%)	Loop with CANDU-like pressure tubes / Forced	Heavy Water / Low Pressure	Thermal	UO2-ThO2 Fuel Rods (Development Required)	LEU-UO2 Rods (Mature)
Intermediate (33%)	Loop with CANDU-like pressure tubes / Forced	Light water and graphite in pressure tubes / High pressure + graphite reflector (and low pressure water ring in wet version)	Thermal	For dry version, TRISO Coated UO2 in a compact in a block of graphite coated with SiC (development required)	For wet version, CANDU type fuel bundle with central rings replaced with a SiC coated graphite plug (development required)
Intermediate (33%)	Loop with CANDU-like pressure tubes / Forced	Light water and graphite in pressure tubes (dry calandria)/ High Pressure	Thermal	Seed: uranium-15%zirconium fuel rods, Blanket: BISO coated Th02 and UCO fuel particles in graphite matrix compacted into pellets (development required)	MOX, DUPIC

Very High (>40%)	Loop (direct cycle) (*1) / Forced (but no re-circulation pumps) (*2)	Light Water / Supercritical Pressure	Thermal (can also be fast)	UO2 Rods (Mature)	MOX Rods (Mature)
Very High (>40%)	Loop (direct cycle) (*1) / Forced	…	Fast	U- and Th-based nitride or metal (Major Development Required)	MOX Rods (Mature)
Very High (41% for the Mark 1 and >44% for the ALX2)	Loop with thermally insulated Zircaloy pressure tubes / Forced	Heavy Water / Low Pressure	Thermal	UO2-ThO2 Fuel Rods (Development Required)	LEU-UO2 Rods (Mature)
Very High (>40%)	Loop (direct cycle) (*1) / Forced	Light Water / Supercritical Pressure	Thermal	Fluidized Bed SiC-PyC-Coated UO2 Particles (Major Development Required)	…
Intermediate (35%)	Integrated in single vessel / Natural	…	Fast	UO2 Rods (Mature)	MOX Rods (Mature)
Intermediate	Integrated in single vessel / Natural	Heavy Water + Light Water / High Pressure	Fast - Epithermal	UO2 rods (Mature)	…
Intermediate (34%)	Loop (direct cycle) (*1) / Forced	…	Fast	ThO2-UO2-PuO2 Rods (Development Required)	Nitride Rods (Major Development Required)
Intermediate (34%)	Loop (direct cycle) (*1) / Forced	…	Fast	MOX rods with dry reprocessing	…

…	…	Light Water or Heavy Water/Boiling or Pressurized	Fast	MOX fuel in tight low-moderation core	…
Intermediate	Loop PWR	Heavy Water/ Pressurized	Fast	MOX fuel in tight low-moderation core with seed and DU blanket regions	…
Intermediate (34%)	Loop (direct cycle) (*1) / Forced	…	Fast	MOX rods with dry reprocessing	UO2 rods (Mature)

Intermediate (30%)	Loop (direct cycle) (*1) / Forced	Light Water / High Pressure	Thermal	Fluidized Bed of SiC-PyC-Coated UO2 Particles (Fabrication processes need development)	…
Very High (up to 45%)	Loop (direct cycle) (*1) / Forced	Light Water / High Pressure	Thermal	Fluidized Bed of SiC-PyC-Coated UO2 Particles (Fabrication processes need development)	…
Intermediate	Integrated in single tube / Forced	Light Water / High Pressure; Graphite or Water Reflector	Thermal	Fluidized Bed of LEU-UO2 Particles (Major Development Required)	…

…	…	Light Water / High Pressure	Thermal	Homogeneous U02-Th02 fuel rods (Development Required)	Micro heterogeneous U02-Th02 fuel rods (Major Development Required)
…	…	Light Water / High Pressure	Thermal	Single bundle with a U-Zr metal fuel seed region and U02-Th02 blanket rods (Major development required for the seed)	Alternate metal fuel seed and thoria-urania blanket assemblies

Intermediate	Loop PWR	Light Water / High Pressure	Thermal	Movable U02 seed region + U02-Th02 blanket region (Development required)	…

…	…	Light Water, Heavy Water	Thermal	Actinides from LWR spent fuel, dry-reprocessed, resintered in MOX pellets and (if re-enriched) used in LWRs, (if not re-enriched) used in CANDU reactors	…

Intermediate	Loop PWR	Light Water / High Pressure	Thermal	Homogeneous MOX rods with varying Pu content	…
Intermediate	Loop PWR	Light Water / High Pressure	Thermal	MOX rods + enriched UO2 rods	…
Intermediate	Loop PWR	Light Water / High Pressure	Thermal	Pu inert matrix annular rods for increased moderation and coolability (Development required) + traditional UO2 rods	…
…	…	Light Water, Heavy Water	Thermal, Fast	Uses Np-enriched MOX fuel to extend fuel lifetime and minimize reactivity swing	…

inition a loop-type reactor.
ıal loop is pumped.
 Very High (>40%)
;).
nsequences of a certain accidents are mitigated by passive means.
ials are well-established; Modest = fuel and/or materials need development and testing; Extensive = fuel and/or materials need signifi

Irradiation Cycle / Refueling (*4)	Cladding Materials (Status)	Reactivity Control	Decay Heat Removal System	Containment
Long / offline	Zircaloy (Mature)	Top-entry CRs, Burnable poisons, No boron	Active (thru the SGs) + Passive (heat exchangers and air and water on the outer containment surface)	Small (HP spherical with suppression pool)
Long / offline	Zircaloy-4 (Mature)	Top-entry CRs, Burnable poisons, No boron	Passive (emergency heat exchangers)	Small (type TBD)
Intermediate / offline	Zircaloy-4 (Mature)	Top-entry CRs, Burnable poisons, Passive scram, No boron	Passive (HP emergency condensers)	Small (with suppression pool)
Long / offline	Zircaloy (Mature)	Not discussed	Passive (depressurization + containment under water)	Small (partially filled with water)
Long / offline	Zircaloy (Mature)	Top-entry CRs, No boron	Passive (not discussed)	Small (partially filled with water)
Long / offline	Zircaloy (Mature)	Top-entry CRs	Passive (not discussed)	Small (partially filled with water)
Long / offline	Zircaloy (Mature)	Bottom-entry CRs, Burnable poisons	Passive (PRISM-style RVACS directly at the vessel outer surface)	No; if needed, small (dry spherical HP)
Long / offline	Need development of cladding material for high burnup (100MWd/kg)	Top-entry CRs, No boron	Passive (air on the outer containment surface)	Small (partially filled with water)
Intermediate (18 months refueling, 4.5yr life for most assemblies) / offline	Zircaloy (Mature)	Top-entry CRs, Boron, Burnable poisons, Passive scram	Passive (LP Emergency Condensers)	Small (entirely filled with pressurized water) + a building to protect against external events
Not discussed	Zircaloy (Mature)	Overmoderated reactor, no CRs, no boron	Not discussed	Not discussed

Long / offline	Zircaloy (Mature)	Top-entry CRs	Passive (depressurization + AP600-like containment)	Large volume BWR/PWR hybrid
Long / offline	Zircaloy (Mature)	Bottom-entry CRs	Passive (depressurization + suppression pool + AP600-like containment)	Small
Short then Long / offline	At first Zircaloy (Mature), then ?	Bottom-entry CRs, Burnable poisons	Active (gas turbine & diesel-driven ECCS) + Passive (in-containment heat exchangers)	Smaller than conventional BWR (with suppression pool)
Long / offline	Zircaloy (Mature)	Bottom-entry CRs	Active (gas turbine & diesel-driven ECCS) + Passive (in-containment heat exchangers)	Small (with suppression pool)
Intermediate / offline	Zircaloy (Mature)	Bottom-entry CRs	Passive (HP Emergency Condensers + LP Containment Cooling Tank)	Large (with suppression pool)

Intermediate / continuous online refueling	Zircaloy (Mature)	Online refueling	Active (traditional CANDU ECCS)	Yes
Intermediate / continuous online refueling	SiC for dry version, ZrC or SiC for wet version (Major development needed)	Scram by flooding the calandria tank, Online refueling	Passive (flooding of the calandria tank for dry version, natural circulation of the water in the annular ring for the wet version)	Yes (Cooled by natural circulation of air on the outside)
Long / continuous online refueling	Zircaloy (Mature)	Scram by flooding the calandria tank, Online refueling	Passive (flooding of the calandria tank)	Yes (cooled by natural circulation of air)

Intermediate (1-3)yrs / Offline	Stainless steel, high nickel alloys, or titanium alloys (Major Development Required)	Top-entry CRs, Burnable poisons	Active (but passive systems can also be used)	Small (with suppression pool)
Long / offline	High nickel alloys or austenitic-martinsitic stainless steels (Major Development Required)	Top-entry CRs, Feed water flow	Active (similar to ABWR)	Small
Intermediate / continuous online refueling	Coated Zr-2.5%Nb for Mark 1, not discussed for ALX2 (Development of high temperature cladding required)	Online refueling	Active (traditional CANDU ECCS)	Yes
Intermediate / continuous online refueling	SiC and pyrocarbons (Major Development Required)	Bottom-entry CRs, Online refueling	Passive (Radial Conduction)	Not Discussed

Long / offline	Not Discussed	Top-entry CRs, Low control requirements from reactivity swing minimization	Passive (HP emergency condensers)	Small
Long / offline	Metal	Top-entry CRs, Low control requirements from reactivity swing minimization	Passive (depressurization + in-containment heat pipes)	Small (with suppression pool)
Long / offline	Not Discussed	Streaming channels for negative void coefficient, Low control requirements from reactivity swing minimization	Not Discussed	Not Discussed
Intermediate / offline	Zircaloy (needs to verify performance in fast spectrum)	Bottom-entry CRs with moderator-displacing follower, Re-circulation pumps, Low control requirements from reactivity swing minimization	Active (diesel-driven ECCS) + Passive (in-containment heat exchangers)	Yes

Intermediate for breeding, Long for internal conversion / offline	Zircaloy or advanced stainless steel	Low control requirements from reactivity swing minimization, Short fuel rod for negative void coefficient	…	…
Long / offline	Stainless steel	Low control requirements from reactivity swing minimization, internal blanket regions for negative void coefficient	…	…
Intermediate / offline	Not Discussed	Streaming channels for negative void coefficient, Low control requirements from reactivity swing minimization	Similar to ABWR	Similar to ABWR

Intermediate / continuous online refueling	SiC-PyC in Water (Development Required)	Bottom-entry CRs, Online refueling	Passive (Radial Conduction)	Not Discussed
Intermediate / continuous online refueling	SiC-PyC in Water (Development Required)	Bottom-entry CRs, Online refueling	Passive (Radial Conduction)	Not Discussed
Not Discussed / continuous on line refueling	Zircaloy (Mature)	Primary coolant flow, Passive scram, No CRs, No boron	Passive (Convection to air or water)	Underground containment

Long / offline	Zircaloy (Mature)	Low control requirements from reactivity swing minimization	…	…
Long / offline	Zircaloy (Mature)	Low control requirements from reactivity swing minimization	…	…

Long / offline	Zircaloy (Mature)	No control rods, reactivity controlled by movable seed rods	…	…

…	Zircaloy (Mature)	Not discussed	…	…
Short / Offline	Zircaloy (Mature)	Pu hardens the spectrum and reduces the worth of the control rods. More control rods required	…	…
Short / Offline	Zircaloy (Mature)	Pu hardens the spectrum and reduces the worth of the control rods. More control rods required	…	…
Short / Offline	Zircaloy (Mature)	Pu hardens the spectrum and reduces the worth of the control rods. More control rods required	…	…
Long / offline	Not Discussed	Low control requirements from reactivity swing minimization	…	…

cant, long-term development and testing.

ICEPTS

Important Safety Characteristics (*5)	Proliferation Characteristics	Resource Utilization	Economic Characteristics	R&D Needs by reactor concept (*6)
LOCAs (large and small) eliminated, LOFAs mitigated	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Minimal R&D costs, nuclear island simplification, factory fabricability	Minimal
Large LOCAs eliminated, LOFAs mitigated; large margins to CHF	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Minimal R&D costs, nuclear island simplification, factory fabricability	Minimal
Large LOCAs, LOFAs, CRD ejection eliminated	Comparable to current LWRs	Comparable to current LWRs	Low power per module, minimal R&D costs, nuclear island simplification, factory fabricability	Minimal
Large LOCAs, LOFAs eliminated	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Very low power per module, low efficiency, minimal R&D costs, nuclear island simplification, factory fabricability	Minimal (with UO2 fuel); Modest (with Th fuel)
Large LOCAs, LOFAs, CRD ejection eliminated	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Very low primary system pressure and efficiency, minimal R&D costs, nuclear island simplification, factory fabricability	Minimal
Large LOCAs, CRD ejection eliminated	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Limited Application, minimal R&D costs, nuclear island simplification, factory fabricability	Minimal
LOCAs (small and large), LOFAs, CRD ejection eliminated; large margins to CHF and instabilities	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Low power per module, minimal R&D costs, nuclear island simplification, factory fabricability	Minimal
Large LOCAs and severe accidents eliminated; Seismic response may be an issue	Low Pu production + dirty Pu isotopics	Th cycle	High burnup, fully automated I&C, maintenance may be difficult with a small, partly water-filled containment	Modest: Th-U metal fuel development, cladding development, passive systems for a large PWR
LOCAs (large and small) eliminated, ATWS and LOFAs mitigated	Comparable to current LWRs	Comparable to current LWRs	Significant plant simplification, however, maintenance may be difficult and the capital costs high with water-filled double wall pipe and vessels, minimal R&D costs	Minimal
Passive shutdown	Comparable to current LWRs	Comparable to current LWRs	Elimination of the control system	Minimal

Comparable to SBWR and AP600 with greater simplicity, severe accidents are included in the design	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs + possibility of Th cycle	Simple design and increased reliance on passive safety systems, improved capacity factors, however, a large containment is required, minimal R&D costs	Minimal (with UO2 fuel); Modest (with Th fuel)
Comparable to SBWR and AP600	Low Pu production + dirty Pu isotopics	Th cycle	Very low power, minimal R&D costs	Modest: Th fuel development
Somewhat safer than SBWR and AP600	Very long irradiation cycle (15 years), replacement of sealed RPV possible	Comparable to current LWRs	Reactor and turbines in one building, no re-circulation pumps, ship hull containment, no fuel pool, reduced primary coolant volume, factory fabrication, short construction time (20 months)	Modest: development of oxide fuel for use well beyond current LWR burnups
Comparable to current BWRs + smaller power density	Comparable to current LWRs + long irradiation cycle	Comparable to current LWRs	Limited application, the total product value may be high due to co-generation, minimal R&D costs	Minimal
Comparable to SBWR and AP600	Comparable to current LWRs	Comparable to current LWRs	Minimal R&D costs	Minimal

Tritium eliminated from the primary circuit + thicker pressure tubes	Low Pu production + dirty Pu isotopics	Th cycle (neutron efficient because of moderation)	Forty percent less capital cost than current CANDUs	Modest: development of high pressure and temperature pressure tubes and modified shutdown system components
Reflooding of the primary system unnecessary, tritium eliminated	Comparable to current LWRs	High burnup and possible reuse of the fuel in CANDUs	Heavy water replaced with light water, boron is eliminated, safety systems are reduced	Extensive fuel development
Reflooding of the primary system unnecessary, tritium eliminated	Low Pu production + dirty Pu isotopics	Th cycle + can burn Pu	Heavy water replaced with light water, boron is eliminated, safety systems are reduced	Extensive fuel development

CHF eliminated, negative void coefficient, but small water inventory and no natural circulation paths	Comparable to current LWRs	High conversion/breeding is achievable more easily than in current LWRs	Very high thermal efficiencies, smaller BOP components and no steam generators or separators, possible material problems, thicker reactor pressure vessel and piping, but less total material	Extensive: cladding and internal structural materials development
CHF eliminated, small water inventory and no natural circulation paths	No Pu production (with a breeding ratio of 1) + dirty Pu isotopics	Can burn actinides from spent LWR fuel (no need to mine uranium)	Very high thermal efficiencies, no steam generators or separators, complicated ECCS, possible material problems, thicker reactor pressure vessel and piping	Extensive: cladding and internal structural materials development, metal and nitride fuels development
CHF eliminated, small water inventory and no natural circulation paths	Low Pu production + dirty Pu isotopics	Th cycle (neutron efficient because of moderation)	Very high thermal efficiencies, smaller BOP components and no steam generators or separators, possible material problems, thicker pressure tubes and piping	Extensive: cladding materials development, development of very high pressure and temperature pressure tubes
CHF eliminated + good FP retention in the fuel + decay heat removed by radiation and conduction	Coated particles hard to reprocess	Comparable to current LWRs	Very high thermal efficiencies, smaller BOP components and no steam generators or separators, possible material problems, thicker reactor pressure vessel and piping, but less total material	Extensive: pebble fuel and structural materials development

Large LOCAs, LOFAs eliminated	Comparable to current FBRs	High conversion U/Pu cycle	Cost of heavy water coolant, heavy water loss	Modest: cladding and internal structure materials, nuclear data, controllability (small beta), negative void coefficients, heavy water decomposition
Large LOCAs, LOFAs eliminated	Comparable to current LWRs + long irradiation cycle	High conversion U/Pu cycle + can burn MA and FPs	Heavy water dilution, relatively high fuel costs, relatively low power, simplified ECCS, reduced number of CRDs, and high capacity factors	Modest: cladding and internal structural materials, nuclear data, negative void coefficients, coolant density wave instabilities, heavy water decomposition
Reactivity swing minimized	Low Pu production + dirty Pu isotopics	Th cycle + can burn Pu	Capital cost similar to ABWR	Modest: cladding materials, nuclear data, controllability (small beta), negative void coefficients, coolant density wave instabilities, CHF margin and accident coolability in tight cores
Fast reactor with negative void response + safety similar to ABWR	Dry reprocessing; no actinides separation	High conversion U/Pu cycle + can burn MA and FPs	Capital cost similar to ABWR	Modest: cladding materials, nuclear data, controllability (small beta), negative void coefficients, coolant density wave instabilities, CHF margin and accident coolability in tight cores

Reactivity swing minimization	Comparable to current FBRs	Can breed Pu or high internal-conversion U/Pu cycle	Requires reprocessing	Modest: development of cladding materials for water cooled fast reactors and confirmation of tight lattice core coolability during accidents
Reactivity swing minimization	Comparable to current FBRs	Can breed Pu or high internal-conversion U/Pu cycle	Requires reprocessing	Modest: development of cladding materials for water cooled fast reactors and confirmation of tight lattice core coolability during accidents
Fast reactor with negative void response + safety similar to ABWR	Dry reprocessing; no actinide separation	High conversion U/Pu cycle + can burn MA and FPs	Capital cost similar to ABWR	Modest: cladding materials, nuclear data, controllability (small beta), negative void coefficients, coolant density wave instabilities, CHF margin and accident coolability in tight cores
Good FP retention in the particle fuel	Coated particles hard to reprocess	Comparable to current LWRs	Somewhat higher plant capacity factors with online refueling	Extensive: pebble fuel reliability in fully-fluidized water bed, safety testing
CHF eliminated + good FP retention in the fuel + decay heat removed by radiation and conduction	Coated particles hard to reprocess	Comparable to current LWRs	Very high plant efficiency, somewhat higher plant capacity factors with online refueling	Extensive: pebble fuel and structural materials development, safety testing
The suspended bed is critical only under certain conditions. Upset conditions lead to a loss of criticality.	Comparable to current LWRs	Comparable to current LWRs	Need a SG, a pump and refueling machine per fuel assembly	Extensive: pebble fuel reliability in fully-fluidized water bed, fuel fabrication technology
Reactivity swing minimization, significant power peaking when micro-hetrogeneous fuel is used	Low Pu production + dirty Pu isotopics	Th cycle + can burn Pu	High SWU costs	Modest: demonstration testing of thoria-urania fuel, Extensive ; safety testing of micro-heteregeneous fuel
Reactivity swing minimization, significant power peaking a beginning-of-cycle	Very low Pu production + dirty Pu isotopics	Th cycle + can burn Pu	High SWU costs, uncertain metal fuel fabrication costs	Extensive: metal driver fuel development and safety testing

Reactivity swing minimization, significant power peaking a beginning-of-cycle	Very low Pu production + dirty Pu isotopics	Th cycle, thermal breeder	Complicated and possibly costly core design	Modest: already successful demonstrated

…	Dry reprocessing; no separation of the actinides from the uranium and most of the fission products	Allows recycle of LWR spent fuel back into a LWR with only a slight addition of fissile material or directly into a CANDU, therefore, there is little or no mining	Disposal of the gaseous and volatile fission products and remote fuel fabrication in a hot cell are expensive	Extensive (the process has never been shown to be feasible on a production scale)

Smaller delayed neutron fraction; total load of Pu must be limited	Comparable to current FBRs	Pu recycling	Requires reprocessing	Minimal: MOX fuel is established
Smaller delayed neutron fraction; total load of Pu must be limited	Comparable to current FBRs	Pu recycling	Requires reprocessing	Minimal: MOX fuel is established
Smaller delayed neutron fraction; total load of Pu must be limited	Comparable to current FBRs	Pu recycling	Requires reprocessing	Extensive: annular rods with Pu and CeO2 need to be developed
Reactivity swing minimization	Dirty Pu Isotopics	High BU + recycling of Pu and Np	Requires reprocessing	Modest: development of Np-enriched MOX fuel that will achieve 200MWd/kg burnup

R&D Needs by group	Estimated Time of Deployment	Group Evaluator
1) Design and accessibility of in-vessel control rod drives. 2) Design and accessibility of the in-vessel HXs. 3) Long-cycle corrosion control of the cladding materials.	<2015	Carelli (Westinghouse), MacDonald (INEEL), Delmastro (CNEA)
1) Accessibility of the underwater primary-system components. 2) Seismic response of the large distributed loop.	<2015 (with U fuel) >2015 (with Th fuel)	Park (KAERI), Lee (DHICO), Lauret (Framatome)

1) Overall R&D needs are minimal because these are mostly small BWRs with little conceptual innovation. 2) Long-cycle corrosion control of the cladding materials.	<2015 (with U fuel) >2015 (with Th fuel)	Devine (Polestar), Schultz (INEEL), Diamond (BNL)

1) Development of higher pressure and temperature pressure tubes. 2) Development of metal and TRISO and BISO coated fuels for LWRs.	CANDU-NG <2015, W5 and W28 >2015	Hedges (AECL), Park (KAERI), Lee (DHICO)

1) Development of fuel, cladding and structural materials. 2) Demonstration of adequate reactor performance and safety. 3) Metal and nitride fuels development (for TWG1)	>2015	Was (U-Michigan), Corradini (U-Wisconsin), Smith (Dominion)

1) Demonstration of the reliability of clad and structural materials in a water-cooled fast reactor. 2) Prevention of CHF or overheating in tight-lattice water-cooled cores.	>2015	Diamond (BNL), Vasile (CEA)

3) Improved nuclear data		

1) Demonstration of the reliability of pebble fuel in fully-fluidized beds. 2) Fabrication of large particles by CVD. 3) Fabrication of Zircaloy clad UO2 spherical fuel	>2015	MacDonald (INEEL)

1) Demonstration of homogeneous U-Th oxide fuel. 2) Development of U-Zr metallic fuel for water-cooled reactors. 3) Demonstration of thermal-hydraulic safety in the high-power-peaking heterogeneous fuel.	>2015	MacDonald (INEEL), Diamond (BNL)

1) Handling of waste streams from dry reprocessing. 2) Demonstration of remote oxide-fuel pellet refabrication in hot cells.	<2015	Hedges (AECL), Park (KAERI)

1) Demonstration of high-Pu core controllability. 2) Development of advanced reprocessing techniques for low-cost, proliferation-resistant multiple recycling of Pu and MA. 3) Development of annular rods for the APA concept.	<2015	Vasile (CEA), Diamond (BNL), MacDonald (INEEL)