Back to Contents

GIF-003-00

Generation IV Roadmap

R&D Scope Report for Water-Cooled Reactor Systems

Issued by the Nuclear Energy Research Advisory Committee and the Generation IV International Forum

December 2002

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CONTENTS

FIGURES

TABLES

EXECUTIVE SUMMARY

This report is one of a series of technical submittals prepared by the Generation IV Technical Working Group on Advanced Water-Cooled Reactor System Concepts (TWG1). Its purpose is to present the TWG1 evaluations and recommendations regarding research and development (R&D) work needed to bring to fruition the water-cooled reactor systems chosen for continued support under the Generation IV Program.

Section 2 of this report, “Concept Description and Final Screening Evaluations,” presents summary descriptions and updated quantitative assessments of the five reactor concept sets that were recommended by TWG1 for further consideration and inclusion in the Generation IV Roadmap. The five reactor concept sets recommended by TWG1 are Integral Primary System Reactors, Simplified Boiling Water Reactors, Pressure Tube Reactors (NG-CANDU), Supercritical Water-Cooled Reactors (Fast and Thermal Spectrum), and High Conversion Water-Cooled Reactors.

After TWG-1’s concept set evaluation and down-selection recommendation, the Generation IV International Forum (GIF) reached agreement on the specific concept sets to be carried forward. The GIF selected concepts are:

- Supercritical Water-Cooled Reactors (SWCR) (Fast and Thermal) – Generation IV candidate
- Integral Primary System Reactors (IPSRs), Pressure Tube Reactors [Next Generation-Canadian Deuterium Uranium Reactor (NG-CANDU)], and High Conversion Advanced Boiling Water Reactors (HCABWRs) – candidates for International Near Term Deployment.

Section 3 of this report, “Technology Gaps and Required R&D,” presents the TWG1 assessment of the R&D work necessary to bring the GIF-selected concept sets from their current stage of development to readiness for detailed engineering and licensing.

Since the SCWR was the only water-cooled reactor system to be included in the proposed Generation IV R&D program, the remainder of this executive summary focuses on that concept set.

SCWR Description

SCWRs are high-temperature, high-pressure water-cooled reactors that operate above the thermodynamic critical point of water (374°C, 22.1 MPa or 705°F, 3208 psia). These systems may have a thermal or fast neutron spectrum, depending upon the core design. SCWRs may have significant advantages compared to state-of-the-art LWRs in the following areas:

- SCWRs attain significant increases in thermal efficiency relative to current generation LWRs. The efficiency of a SCWR can approach 45%, compared to 33–35% for LWRs.
- A lower coolant mass flow rate per unit core thermal power results from the higher enthalpy content of the coolant. This leads to a reduction in the size of the reactor coolant pumps, piping, and associated equipment, and a reduction in the pumping power.
- A lower coolant mass inventory results from the once-through coolant path in the reactor vessel and the lower coolant density. This opens the possibility of smaller containment buildings.
- No boiling crisis (i.e., departure from nucleate boiling or dry out) exists during normal operation due to the lack of a second phase in the reactor, thereby avoiding discontinuous heat transfer regimes within the core.
- Elimination of steam dryers, steam separators, re-circulation pumps, as well as steam generators. Therefore, the SCWR will be a simpler plant with fewer major components.

The Japanese supercritical light water reactor (SCLWR) with a thermal spectrum has been the subject of the most development work in the last 10 to 15 years and is the reference concept. The SCLWR reactor vessel is similar in design to a PWR vessel (although primary coolant system is a direct-cycle, BWR-type system). High-pressure (25.0 MPa) 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. This strategy is employed to provide moderation in the core. The coolant is heated to about 510°C and delivered to a power conversion cycle, which blends LWR and supercritical fossil plant technology: high-, intermediate- and low-pressure turbines are employed with two reheat cycles. The overnight capital cost for a 1700-MWe SCLWR plant may be as low as $700/kWe (about half that of current ALWR capital costs), considering the effects of simplification, compactness, and the economy of scale. The operating costs may be 35% less than current LWRs.

Two variants of the thermal design were also qualitatively assessed: a heavy water-moderated thermal reactor within CANDU type pressure tubes and a thermal reactor with small (2 to 5 mm) spherical fuel pebbles with TRISO coatings.

The SCWR can also be designed to operate as a fast reactor. The difference between thermal and fast versions is primarily the amount of moderator material in the SCWR core. The fast spectrum reactors use no additional moderator material, while the thermal spectrum reactors need additional moderator material in the core. Both water and solid moderator rods are feasible.

Technology Base for the SCWR

Much of the technology base for the SCWR can be found in the existing LWRs as well as commercial supercritical water-cooled fossil-fired power plants. However, there are some relatively immature areas. There have been no prototype SCWRs built and tested. For the reactor primary system, there has been very little in-pile research done on potential SCWR materials or designs, although there has been some SCWR in-pile research done for defense programs in Russia and the United States. Limited design analysis has been underway over the last 10 to 15 years in Japan, Canada, and Russia. For the balance of plant, there has been development of turbine generators, piping, and other equipment that has been extensively used in supercritical water-cooled fossil-fired power plants. The SCWR may have some success at adopting portions of this technology base.

Technology Gaps for the SCWR

The important SCW technology gaps are in the areas of:

- SCWR materials and structures, including:
  - Corrosion and stress corrosion cracking (SCC)
  - Radiolysis and water chemistry
  - Dimensional and microstructural stability
  - Strength, embrittlement, and creep resistance
- SCWR safety, including power-flow stability during operation
- SCWR plant design.

Important viability issues are found within the first two areas, and performance issues are found primarily within the first and third areas.

SCWR Fuels and Materials R&D

The SCW environment is unique and few data exist on the behavior of materials in SCW under irradiation and in the temperature and pressure ranges of interest. At present, no candidate alloy has been confirmed for use as either the cladding or structural material in thermal or fast spectrum SCWRs. Potential candidates include austenitic stainless steels, solid solution and precipitation hardened alloys, ferritic-martensitic alloys, and oxide dispersion-strengthened alloys.

The fast SCWR design would result in greater doses to cladding and structural materials than in the thermal design by a factor of 5 or more. The maximum doses for the core internals are in the 10–30 dpa range in the thermal design, and could reach 100–150 dpa in the fast design. These doses will result in greater demands on the structural materials in terms of the need for irradiation stability and effects of irradiation on embrittlement, creep, corrosion, and SCC. The generation of helium by transmutation of nickel is also an important consideration in both the thermal and fast designs because it can lead to swelling and embrittlement at high temperatures. The data obtained during LMFBR development will play an important role in this area.

To meet these challenges, the R&D plan for the cladding and structural materials in the SCWRs focuses on acquiring data and a mechanistic understanding related to the following key property needs: corrosion and SCC, radiolysis and water chemistry, dimensional and micro-structural stability, and strength and creep resistance.

Corrosion and SCC

The SCWR corrosion and SCC research program should focus on obtaining the following information:

- Corrosion rates in SCW at temperatures between 280 and 620°C. The corrosion should be measured under a wide range of oxygen and hydrogen contents to reflect the extremes in dissolved gasses
- Composition and structure of the corrosion films as a function of temperature and dissolved gasses
- The effects of irradiation on corrosion as a function of dose, temperature, and water chemistry
- SCC as a function of temperature, dissolved gasses, and water chemistry
- The effects of irradiation on SCC as a function of dose, temperature, and water chemistry.

The corrosion and SCC R&D program will be organized into three parts: an extensive series of out-of-pile corrosion and SCC experiments on unirradiated alloys, companion out-of-pile corrosion and SCC experiments on irradiated alloys, and in-pile loop corrosion and SCC tests. It is envisioned that at least two and maybe as many as four out-of-pile test loops would be used, some addressing the corrosion issues and others addressing the SCC issues. At least two such loops should be built inside a hot cell in order to study preirradiated material. Facilities to preirradiate samples prior to corrosion and SCC testing will be required. Doses of 10–30 dpa will be required to support the thermal design and doses into the 100–150 dpa range will be required to support the fast reactor version of the design. This work should be carried out over a 6–10 year time span for unirradiated materials and the same for irradiated materials. Accelerators capable of producing high currents of light ions may also be utilized to study irradiation effects on corrosion and SCC in a postirradiation mode at substantially lower cost than reactor irradiations.

About mid-way through the out-of-pile work, at least one, and more likely two, in-pile test loops should start operating under both fast and thermal spectrum irradiation conditions (a total of 3 to 4 loops). The in-pile loops will be used to study corrosion, SCC, and water chemistry control issues (see below). We will probably need about 10 years of in-pile testing in these loops to obtain all data required to support both the viability and performance phases of the development of the thermal spectrum version of the SCWR and maybe as much as 15 years of testing to obtain the needed information for the fast spectrum SCWR. A postirradiation characterization and analysis program will accompany the reactor- and accelerator-based irradiations beginning in year 5 and extending for a 10-year period.

Radiolysis and Water Chemistry

The SCWR water chemistry research program should be focused on obtaining the following information:

- The complete radiolysis mechanism in SCW as a function of temperature and fluid density, including the effects of radiation on the radiolysis yields
- The chemical potential of H_2, H_2O_2, O_2, and various radicals in SCW over a range of temperatures (280–620°C)
- Recombination rates of various radicals, H_2, H_2O_2, and O_2 in SCW over a range of temperatures (280–620°C)
- Formation and reaction of other species by radiolytic processes
- Effectiveness of hydrogen and noble metal water chemistry on suppressing the electrochemical corrosion potential of the SCW.

Two research avenues are envisioned to obtain this information. First, beam ports and accelerators can be used to irradiate SCW chemistries and study the characteristics of the recombination processes in some detail. This information will be integrated into a model of the water radiolysis mechanism. Second, water chemistry-control studies can be performed using the in-pile test loops needed for the corrosion and SCC research discussed above.

Dimensional and Microstructural Stability

The SCWR dimensional and microstructural research program should focus on obtaining the following information:

- Void nucleation and growth and the effect of He production on void stability and growth and He bubble nucleation and growth as a function of dose and temperature
- Development of the dislocation and precipitate microstructure and radiation-induced segregation as a function of dose and temperature
- Knowledge of irradiation growth or irradiation-induced distortion as a function of dose and temperature
- Knowledge of irradiation-induced stress relaxation as function of tension, stress, material, and dose.

While many of the test specimens for this work will be irradiated in the corrosion and SCC in-pile loops discussed above, accelerator-based irradiation offers a rapid and low-cost alternative to the handling and analysis of neutron-irradiated material. Much of the needed information will be obtained during postirradiation examinations over the 15-year period of the corrosion and SCC tests. In addition, some stand-alone capsule irradiation tests in test reactors should be performed in order to obtain scoping data on a range of candidate materials in a timely manner. It may be possible to utilize some existing LMFBR data in this research.

Strength, Embrittlement, and Creep Resistance

The SCWR strength, embrittlement, and creep resistance research program should be focused on obtaining the following information:

- Tensile properties as a function of dose and temperature
- Creep rates and creep rupture mechanisms as a function of stress, dose, and temperature
- Creep-fatigue as a function of loading frequency, dose, and temperature
- Time dependence of plasticity and high-temperature plasticity
- Fracture toughness as a function of irradiation temperature and dose
- Ductile-to-brittle transition temperature (DBTT) and helium embrittlement as a function of dose and irradiation temperature
- Changes in microstructure and mechanical properties following design basis accidents.

The research program will be aimed at high-temperature performance of both irradiated and un-irradiated alloys and also at low-temperature performance of irradiated alloys. High-temperature testing will include yield property determination, time-dependent (creep) experiments and also the effect of fatigue loading with a high mean stress. This program will be conducted first on unirradiated alloys over a period of 8 years. Midway through the program, testing will begin on irradiated materials for a period of 10 years. The low temperature fracture toughness/DBTT program will require 10 years.

SCWR Safety R&D

A SCWR safety research program is envisioned, organized around the following topics:

- Reduced uncertainty in SCW thermal-hydraulic transport properties.
- Further development of appropriate fuel cladding to coolant heat transfer correlations for SCWRs under a range of fuel rod geometries.
- SCW critical flow measurements, as well as models and correlations.
- Measurement of integral loss-of-coolant accident (LOCA) thermal-hydraulic phenomena in SCWRs and related computer code validation.
- Fuel rod cladding ballooning during LOCAs.
- SCWR design optimization studies including investigations to establish the effectiveness of passive safety systems.
- Power-flow stability assessments.

The purpose of making additional basic thermal-hydraulic property measurements at and near the pseudo-critical temperatures would be to improve the accuracy of the international steam-water property tables. This work could be done over a 3–5-year time frame.

The fuel cladding-to-SCW heat transfer research should consist of a variety of out-of-pile experiments starting with tubes and progressing to small and then relative large bundles of fuel rods. The bundle tests should include some variations in geometry (such as fuel rod diameter and pitch, bundle length, channel boxes), axial power profiles, coolant velocity, pressure, and grid spacer design. The larger bundle tests will require megawatts of power and the ability to design electrically heated test rods with appropriate power shapes. It is expected that this program might take 5–6 years.

The SCW critical flow experiments would be out-of-pile experiments with variations in hole geometry and water inventory. This research would take 4–5 years.

The integral SCWR LOCA thermal-hydraulic experiments would be similar to the Semiscale experiments previously conducted for the U.S. Nuclear Regulatory Commission to investigate LOCA phenomena for the current LWRs. A test program and the related computer code development would take about 10 years. It may be possible to design this facility to accommodate the heat transfer research discussed above, as well as the needed LOCA testing, and even some thermal-hydraulic instability testing.

Fuel rod cladding ballooning is an important phenomenon that may occur during a rapid depressurization. Although considerable work has been done to measure and model the ballooning of Zircaloy clad fuel rods during LOCAs, little is known about the ballooning behavior of austenitic or ferritic-martensitic stainless steel or nickel-based alloy clad fuel rods during a LOCA. It is expected that this information could be obtained from out-of-pile experiments using fuel rod simulators. The research would take 4–6 years.

All of the known accident scenarios must be carefully evaluated (large- and small-break LOCAs, RIAs, loss of flow, main steam isolation valve closure, over-cooling events, anticipated transients without scram, and high- and low-pressure boil off) to ensure compliance with reactor protective criteria. There may be safety features (such as redundant reactivity shout down systems) that require special designs. It is estimated that tests can be conducted within a period of 3–5 years.

The objective of the power-flow stability R&D is better understanding of neutronic-thermal-hydraulic instability phenomena (including side-to-side instabilities) in SCWRs, the identification of the important variables affecting these phenomena, and ultimately the generation of maps identifying the stable operating conditions of the different SCWRs designs. Consistent with the U.S. Nuclear Regulatory Commission approach to BWR licensing, the licensing of SCWRs will probably require, at a minimum, demonstration of the ability to predict the onset of instabilities. This can be done by means of a frequency-domain linear analysis.

Both analytical and experimental studies need to be carried out for the conditions expected during the different operational modes and accidents. The analytical studies can obviously be more extensive and cover both works in the frequency domain as well as direct simulations. These studies can consider the effect of important variables such as axial and radial power profile, moderator density and fuel temperature reactivity feedback, fuel rod thermal characteristics, coolant channel hydraulic characteristics, heat transfer phenomena, and core boundary conditions. Mitigating effects like orificing, insertion of control rods, and fuel modifications to obtain appropriate thermal and/or neutronic response time constants can also be assessed using analytical simulations. It is envisioned that instability experiments will be conducted at the multipurpose SCW thermal-hydraulic facility planned for the safety experimentation discussed above. The test section will be designed to accommodate a single bundle as well as multiple bundles. This will enable studying in-phase and out-of-phase density-wave oscillations. Moreover, the facility will provide a natural circulation flow path for the coolant to study buoyancy loop instabilities. It is projected that the instability experiments and related analytical work will require 3 to 4 years. Further work would depend on the issues uncovered during the experimental program.

SCWR Plant Design R&D

Many of the major systems that can potentially be used in a SCWR were developed for the current BWRs, PWRs, and SCW fossil plants. Therefore, the major plant design and development needs that are unique for SCWRs are primarily found in their design optimization, as well as their performance and reliability assurance under SCWR neutronic and thermo-hydraulic conditions. Two major differences in conditions are the stresses due to the high SCWR operating pressure (25 MPa) and the large coolant temperature and density change (approximately 280 to 500°C or more, 800 to 80 kg/m^3, respectively) along the core under the radiation field.

Examples of design features that need to be optimized to achieve competitiveness in economics without sacrificing safety or reliability include the fuel assemblies, control rod drive system, internals, reactor vessel, pressure relief values, coolant cleanup system, reactor control logic, turbine configuration, re-heaters, deaerator, start-up system and procedures, in-core sensors, and containment building. This work is expected to take about 8 to 10 years.

SCWR Balance of Plant R&D

The SCWRs will utilize the existing technology from the secondary side of the supercritical water-cooled fossil-fired plants. Significant research in this area is not needed.

SCWR Fuel Cycle R&D

The thermal spectrum SCWR option will use conventional LEU fuel. The fuel itself is fully developed; however, new cladding materials and fuel bundle designs will be needed. The designs for the thermal spectrum SCWR will need significant additional moderator, i.e., water rods or solid moderation. The designs for the fast spectrum SCWRs will require a tight pitch, but high neutron leakage to create a negative density coefficient. The fast spectrum SCWR option will use mixed plutonium-uranium oxide fuel with advanced aqueous reprocessing. The research needed for this fuel cycle technology is discussed in the Crosscutting Fuel Cycle R&D reports.

ACRONYMS

ABWR	advanced boiling water reactor
AECL	Atomic Energy of Canada Limited
ADS	automatic depressurization system
ALWR	advanced light water reactor
ATR	Advanced Test Reactor
BWR	boiling water reactor
CANDU	Canadian Deuterium Uranium Reactor
CAREM	Central ARgentina de Elementos Modulares
CERT	constant extension rate tension
CHF	critical heat flux
CRDM	control rod drive mechanism
CVD	chemical vapor deposition
DBA	design basis accident
DBTT	ductile-to-brittle-transition-temperature
DOE	Department of Energy
ECCS	Emergency Core Cooling System
EMG	Evaluation Methodology Group
ESBWR	European Simplified Boiling Water Reactor
GDCS	Gravity Driven Core-Cooling System
GE	General Electric
GIF	Generation IV International Forum
HCABWR-II	High Conversion Advanced Boiling Water Reactor-II
HCABWRs	High Conversion Advanced Boiling Water Reactors
HCBWRs	High Conversion Boiling Water Reactors
HCLWR	High Conversion Light Water Reactor
HCPWR	High Conversion Pressurized Water Reactor
IASCC	irradiation assisted stress corrosion cracking
IC	isolation condenser
IGSCC	intergranular stress corrosion cracking
INEEL	Idaho National Engineering and Environmental Laboratory
IPPE	Institute of Physics and Power Engineering
IPSR	Integral Primary System Reactor
IRIS	International Reactor Innovative and Secure
ISPWR	Integral System Pressurized Water Reactor
JAERI	Japan Atomic Energy Research Institute
KAERI	Korean Atomic Energy Research Institute
LEU	low enriched uranium
LOCA	loss-of-coolant accident
LSBWR	Long Operating Cycle Simplified BWR
LWR	light water reactor
MASLWR	Multi-Application Small Light Water Reactor
MIT	Massachusetts Institute of Technology
MOX	mixed uranium-plutonium oxide
MRX	Advanced Integral-type Marine Reactor X
NERI	Nuclear Energy Research Initiative
NG-CANDU	Next Generation-Canadian Deuterium Uranium Reactor
NRC	Nuclear Regulatory Commission

NRU	National Research Universal
O&M	operations and maintenance
ODS	oxide-dispersion strengthened
PCMI	pellet-cladding mechanical interaction
PCCS	passive containment cooling system
PNNL	Pacific Northwest National Laboratory
PSA	probabilistic safety analysis
PSRD	Passively Safe Small Reactor for Distributed Energy System
PWR	pressurized water reactor
R&D	Research & Development
RCIC	Reactor Core Isolation Cooling System
RFI	Request for Information
RHR	residual heat removal
RIA	reactivity-initiated accident
RIS	radiation induced segregation
RMWR	Reduced-Moderation Water Reactor
SBWR	Simplified Boiling Water Reactor
SCC	stress corrosion cracking
SCLWR	supercritical light water reactor
SCW	supercritical water
SEU	slightly enriched uranium
SMART	System-Integrated Modular Advanced Reactor, a small intergral PWR design from Korea
SSBWR	Safe and Simplified Boiling Water Reactor
TWG1	Technical Working Group on Advanced Water-Cooled Reactor System Concepts

R&D Scope Report for Water-Cooled Reactor Systems

1. INTRODUCTION

This report is one of a series of technical submittals prepared by the Generation IV Technical Working Group on Advanced Water-Cooled Reactor System Concepts (TWG1). It presents the TWG1 evaluations and recommendations regarding research and development (R&D) work needed to bring to fruition the water-cooled reactor systems chosen for continued support under the Generation IV Program. This information is presented in two major sections:

- Section 2, "Concept Description and Final Screening Evaluations," presents summary descriptions of the five reactor concept sets selected by TWG1 for further consideration and selection by the Generation IV Roadmap, and updated quantitative assessments of those concept sets.
- Section 3, "Technology Gaps and Required R&D," presents the TWG1 assessment of the R&D work necessary to bring several of those selected concept sets from their current stage of development to readiness for detailed engineering and licensing.

Note that Section 3 does not provide R&D needs or recommendations for all five of the concepts presented in Section 2. Subsequent to TWG-1's concept set evaluation and recommended down-selection, the Generation IV International Forum (GIF) reached agreement on the specific concept sets to be carried forward. TWG 1 has therefore limited its R&D scope recommendations to the GIF-selected concepts. The GIF selected concepts are:

- Supercritical Water-Cooled Reactors (SWCR) (Fast and Thermal) – Generation IV candidate.
- Integral Primary System Reactors (IPSRs), Pressure Tube Reactors [Next Generation-Canadian Deuterium Uranium Reactor (NG-CANDU)], and High Conversion Advanced Boiling Water Reactors (HCABWRs)– candidates for International Near Term Deployment.

1.1 Background

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.

The Generation IV Roadmap program has been organized into four technical working groups, arranged by reactor coolant type: water, gas, liquid metal, and nonclassical. Other cross-cutting working groups are established to provide input in areas that bridge the four basic reactor system technologies (e.g., fuel cycle, economics, reactor system safety, energy production). Membership in these groups includes U.S. and international experts from industry, government, and academia.

Within the Generation IV Program, TWG1 was charged with identifying and evaluating advanced water-cooled-reactor nuclear energy system concepts. The initial activity, as described in our first report titled *Description and Evaluation of Candidate Water-Cooled Reactor Systems,"*(Document 3 on this CD-ROM) was the assessment and "screening for potential" of candidate systems, to provide a sound basis for subsequent additional evaluations, comparisons with other (nonwater) reactor concepts, and final selection of concepts and technology for R&D support.

Advanced water-cooled-reactor nuclear energy system concepts were identified via a formal Department of Energy (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, TWG1 itself collected information on eight concepts, yielding a total of 38 concepts for evaluation.

TWG1 consolidated all but 1 of the 38 reactor and fuel cycle concepts into 10 distinct ***concept sets***, based on their key common features. 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 and was not considered further. TWG1 then conducted a qualitative evaluation of the 10 sets in order to determine their potential to achieve the Generation IV goals. The results of that assessment are published in our first report. The qualitative evaluation was used as a foundation for TWG1 to choose six concepts or concept sets for a quantitative assessment using the January 3, 2002 guidance and scoring sheets, prepared by the Generation IV Roadmap Evaluation Methodology Group (EMG).

The original 10 concept sets are as follows:

1. *Integral Primary System Reactors.* These light water reactor (LWR) 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 (PWRs).* These are modified loop-type 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.* These are 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 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 on the specific design. Both light water and heavy water moderation have been proposed. Plant efficiencies between 40 and 45% can be obtained with use of supercritical water (SCW).
6. *High-Conversion Water-Cooled Reactors.* 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

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.

positive void coefficient is reduced by the use of neutron streaming assemblies and pancake-type cores.

7. *Pebble Fuel Reactors.* 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 (ALWRs) with Thorium/Uranium Fuel.* These are 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 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 unuseable in a nuclear weapon.

10. *ALWRs with Plutonium and Minor Actinide Multi-Recycling.* These are ALWRs with either normal moderation or reduced moderation cores that burn plutonium and minor actinides. Multirecycling 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.

1.2 Rationale and Approach to Down-Selection

TWG1's approach in evaluating the ten concept sets included the following judgments regarding set content and evaluation bases:

- In the case of the IPSRs, the quantitative assessment represents the potential (and development costs) of the entire set, but the scoring relies heavily on information provided TWG1 about the International Reactor Innovative and Secure (IRIS) concept being developed by the Westinghouse Electric Co. and an international group of about 20 industrial firms, universities, and national laboratories (W18).

- The quantitative assessment of the SBWRs is solely focused on the European Simplified Boiling Water Reactor (ESBWR) being developed by the General Electric (GE) Co. (W13). TWG1 decided to focus on the ESBWR because we had much more design information and better cost estimates for that concept than we had for any of the small modular SBWRs.

- The quantitative assessment of the Pressure-Tube Reactors is for the NG-CANDU plant being designed and marketed by Atomic Energy of Canada Limited (AECL) (W6). TWG1 found less potential in the other pressure tube concepts that were submitted.

- Six SCW-cooled concepts were submitted. TWG1 felt that at least two should be subjected to a quantitative assessment, the thermal spectrum SCWR design being developed in Japan at the University of Tokyo with government funding (W21) and a similar reactor with a fast spectrum core (TWG1-1). However, note that the SCWR design being assessed in both cases is not exactly the Japanese design, because TWG1 felt that many of the SBWR passive safety features could and should be added to that design.

- TWG1 was also interested in the CANDU and Pebble-Bed SCW reactor designs, but felt that their potential could be represented by the SCWR-Thermal quantitative assessment.

- In the case of the High Conversion Water-Cooled Reactors, TWG1 decided to do a quantitative assessment on the High Conversion Advanced Boiling Water Reactor (HCABWR)-II design being developed by Hitachi (W9). It was decided not to consider the high conversion PWR (HCPWR) concepts because of concerns about the potential costs of using heavy water in a PWR system. For much the same reason, the Hitachi safe and simplified boiling water reactor (SSBWR) design, with heavy water being diluted with light water during each fuel cycle, was not considered. Among the high conversion boiling water (HCBWR) concepts, it was felt that: (a) the SBWR approach should be eliminated because not all their safety features would be appropriate for a high conversion design with its very tight lattice and, (b) the ABWR-II provides significant advantages over the ABWRs in safety and reliability and economics.

The five concept sets not subjected to a quantitative assessment and, therefore, not being considered further by TWG1 are the Loop PWRs, Pebble Fuel Reactors, ALWRs with Thorium/Uranium Fuel, Advanced Water-Cooled Reactors with Dry Recycling of Spent LWR Fuel, and ALWRs with Plutonium and Minor Actinide Multi-Recycling. The rationale underlying the TWG-1 downselect judgments in these cases is as follows:

- The Loop PWRs were not considered for quantitative assessment because TWG1 felt that the economics of adding what was essentially a second containment vessel would be prohibitive. Also, the plant safety improvement made possible by that approach can be achieved with a number of the other concepts TWG1 chose to consider. And, TWG1 has significant concerns about the practicality of the maintenance of such systems.
- The Pebble Fuel Reactors were not subject to a quantitative assessment because TWG1 felt that their claimed safety advantages could be achieved more readily with other water-cooled concepts that TWG1 chose to consider. In addition, TWG1 was not provided very much design and analysis information to support the claims made by the proponents of those concepts, and it was, therefore, hard to properly judge their viability.
- The ALWRs with Thorium/Uranium Fuel concept set was not subjected to a quantitative assessment because of the poor economics of thorium fuel cycles. Recycle of U-233 is not economic in today's environment of low uranium ore and separative work unit prices. Use of mixed uranium-thorium fuels (either homogeneous or seed and blanket) is not economic because of the relatively high U-235 enrichments needed. Also, the Fuel Cycle Crosscut Group will issue a separate evaluation that will address thorium fuel cycles.
- The Advanced Water-Cooled Reactors with Dry Recycling of Spent LWR Fuel concept set was essentially added to the Pressure Tube Reactor concept set and a quantitative assessment of the NG-CANDU with a DUPIC fuel cycle was prepared. However, that quantitative assessment is not included in this report at the direction of the Fuel Cycle Crosscut Group and the Roadmap management. Other uses (recycle into thermal spectrum reactors) for the dry fuel recycle fuel cycle were not considered by TWG1 because of economic considerations.
- The ALWRs with Plutonium and Minor Actinide Multi-Recycling concept set was not subjected to a quantitative assessment because TWG1 felt that fast spectrum reactors are better suited for actinide management than thermal spectrum reactors. Also, the Fuel Cycle Crosscut Group will issue a separate evaluation that will address these fuel cycles as well.

Since mid-2001, TWG1 has been evaluating reactor system concepts using the quantitative evaluation process developed by the EMG. This process compiles specific numerical scores in the form of probabilistic distributions for each candidate system in each of 28 areas. The results are compiled into goal-area (Sustainability, Reliability and Safety, and Economics) and total composite scores.

The water-cooled reactor system concept scores have been developed and refined through numerous iterations, initially within TWG1 and later including interaction with the crosscutting groups and the other TWGs. The results were finalized following the TWG Co-chairs meeting in Houston in March 2002.

Notwithstanding the process limitations outlined below, these scored results reflect and provide documentation of as thorough an evaluation of the candidate system concepts as is possible with the information currently available.

1.3 Recommended Water-Cooled Reactor Systems

The water-cooled reactor system concepts recommended by TWG1 for continuing consideration for inclusion in the roadmap, and therefore presented in this report, are as follows:

Concept	Section
Integral primary system reactors	2.1
Simplified Boiling Water Reactors	2.2
Pressure Tube Reactors (NG-CANDU)	2.3
Supercritical Water-Cooled Reactors (Fast and Thermal Spectrum)	2.4
High Conversion Water-Cooled Reactors	2.5

This report summarizes the recommended water-cooled reactor system concepts and their R&D needs, including the following:

- A summary of each concept (Section 2), including
  - Concept Set Description
  - Concept Strengths and Weaknesses in Sustainability
  - Concept Strengths and Weaknesses in Reliability and Safety
  - Concept Strengths and Weaknesses in Economics
- Identification of technology gaps, required R&D, and R&D challenges attendant to the selection of these concepts for inclusion in the roadmap (Section 3).
- Individual quantitative assessment score sheets (included on this CD).

1.4 Application of the Quantitative Evaluation Process

The Generation IV quantitative review process has been a key tool in the evaluation and selection of reactor system concepts. It permits methodical, comprehensive, and consistent assessment of concepts and produces a meaningful basis for comparison of the prospective capabilities of each to meet the Generation IV goals and criteria. For those reasons, TWG1 applied a great deal of effort into the quantitative evaluations, and their results strongly influenced the TWG1 selections.

Despite the evident value of the quantitative process, it has inherent limitations that must also be recognized. In many cases, and particularly for reactor systems in the early stages of conceptual development (and for that reason, may be of high importance to Generation IV) there is very high uncertainty in concept potential in meeting many of the criteria. In some instances, concept capability is simply unknowable at this stage. Numerical scoring therefore becomes a matter of judgment, and there has been, in many cases, significant wide variations among the TWG1 members' views.

The process of scoring the concepts has been thought provoking and illuminating, and the score justifications provide useful documentation of the TWG1 consensus judgments. But the composite numerical values should not be considered definitive—they are indicators of potential merit, at best.

2. CONCEPT DESCRIPTIONS AND FINAL SCREENING EVALUATIONS

2.1 Integral Primary System Reactors

2.1.1 Concept Set Description

Seven reactor concepts (see Table 1) were submitted to the DOE-NE RFI that are characterized by a primary system 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 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. Most of the IPSRs are designed for relatively high burnup, long fuel cycles. Some of the concepts are also designed for MOX fuels and/or thorium-based fuels.

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
W18 (IRIS)	Carelli (Westinghouse, USA)	335 MWe	Pressurized, 15.5 MPa	Forced	HP spherical with suppression pool
W10 (SMART)	Chang (KAERI, 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
W25 ("Daisy")	Buongiorno (INEEL, USA)	50-100 MWe	Boiling, 7.4 MPa	Natural	HP spherical, dry
W26 (MASLWR)	Modro (INEEL, USA)	35 MWe	Pressurized, with some boiling, 10.5 MPa	Natural	Partially filled with water

* *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 emphasis in this class of reactors 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.

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 LWR accident initiators by design rather than having to mitigate the consequences of those accidents. For example, large loss-of-coolant accidents (LOCAs) are physically impossible, and 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 MWth (~335 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 335-MWe or less plant size is more appropriate than a large plant.

Five of the concepts (IRIS, CAREM, the System-integrated Modular Advanced Reactor or SMART, the Passively Safe Small Reactor for Distributed Energy System or PSRD, and the Advanced Integral-type Marine Reactor X or MRX), while varying somewhat in size, share a similar design and configuration with modular helical steam generators arranged in an annulus above the core, and therefore were grouped in a single set. A very important characteristic of this set is the comparatively advanced state of development. IRIS is scheduled to start its licensing preapplication in mid-2002; CAREM design has been completed, along with an extensive set of qualification testing, a 35-MWe prototype has been readied for construction, but put on indefinite hold by Argentina's economic conditions; construction of a SMART prototype has been authorized and is expected to start in 2002; and MRX has conducted basic testing for over a decade.

The other two concepts (Multi-Application Small LWR or MASLWR and Daisy) are designated as *trailers*, and are not scored. A *trailer* is a concept with enough dissimilarity to make it different from the chosen set, but also with enough similarities to benefit from the design and R&D effort spent on the concept set. Differences in MASLWR are the lower operating temperature and pressure, which allows for use of a relatively inexpensive turbo-generator (at the expense of a reduced thermal efficiency), the configuration with a single helical-coil steam generator enveloping the region directly above the core, the allowance of significant subcooled boiling in the core, and use of a small cylindrical containment (4 m OD, 18 m high) completely submerged in a large water pool. Daisy is an indirect-cycle BWR that achieves 100% natural circulation, even at relatively high power. With respect to the mainstream IPSR design, its potential advantages include the reduction of the energy stored in the primary system (and consequently the reduction of the containment volume), elimination of the pumps, and the possibility of much more compact steam generators (for given power transferred) with condensing steam on the primary side and boiling water on the secondary side. However, at this time the design has not progressed beyond the initial stage of qualitative concept definition.

The following key assumptions were included in the IPSR quantitative assessment prepared by TWG1:

- The IPSR employs relatively standard low enriched uranium (LEU) dioxide fuel. The U-235 enrichment is 6.7%, the average discharge burnup is 80 MWd/kgHM, and the thermal efficiency is 34.5%. These values were calculated for the CAREM and IRIS high burnup cores with fuel shuffling.
- The hybrid high-pressure/pressure suppression containment of IRIS and SMART was adopted as representative of the concept set. While other concepts share most of its features, the IRIS and SMART containments utilize at their best the safety potential of the integral reactor concept.
- Control rods are driven by internal control rod drive mechanisms (CRDMs). This solution, which prevents rod ejection accidents, is the reference for CAREM and MRX and will also be adopted by IRIS in the longer term.

The quantitative assessment score sheet for the IPSRs and the overall rollup can be found in the score sheet appendix on this CD.

2.1.2 Concept Strengths and Weaknesses in Sustainability

Standard LEU fuel was used in the evaluation of this concept. The U-235 enrichment is 6.7%; the average discharge burnup is 80 MWd/kg_{HM}; and the thermal efficiency is 34.5%. These values were calculated for the CAREM and IRIS high-burnup cores with fuel shuffling. The integral configuration provides space for fuel rod designs with large fission gas plenums, which, along with well-moderated cores, makes the high burnup feasible.

The higher burnup reduces the mass and volume of spent fuel per unit energy generated, while the uranium utilization, the long-term heat output and the radiotoxicity of the IPSR fuel per unit energy generated are judged to be essentially similar to current LWRs. The attractiveness of the spent fuel for development of nuclear weapons is reduced, because the plutonium composition at discharge is rich in even-number nonfissile isotopes. Also, the long in-reactor lifetime of the fuel minimizes the opportunity for diversion at the plant.

The IPSRs also provide somewhat better resistance to sabotage because they make extensive use of multiple and independent passive safety systems, which are hard to interfere with and disable from outside the containment building.

Overall, these reactors are judged to provide a modest improvement over current LWRs in the area of sustainability. The sustainability summary score for the IPSRs is 0.13.

2.1.3 Concept Strengths and Weaknesses in Reliability and Safety

The simplified design of the IPSRs with elimination of many components and drastic simplification of the safety systems has potential for reducing the forced outage rate significantly. Also, advanced diagnostic systems will be used to improve the forced outage rate and minimize routine maintenance. However, the reliability of the innovative in-vessel components cannot be evaluated at this point. In general, there is concern about keeping everything enclosed inside the reactor pressure vessel where the inspections may be less effective. Also, outage rates are, in general, driven by external events, e.g., problems in the turbine building and human error, which are not much affected by any Generation IV design. Therefore, it was judged that on average the IPSRs would be similar to reference in the area of reliability, with the potential of being both better and worse than reference.

On the other hand, the simplified plant design and use of extensive diagnostics has the potential to significantly reduce worker exposures. Also, the internal shields make the outside surface of the vessel essentially nonradioactive, and steam generator inspections can normally be done without opening the reactor pressure vessel, thus resulting in lower doses to do inspection. Further, there is no soluble boron, and refueling is infrequent, all of which also contribute to lower routine exposure risk for workers and probability of accidents.

Reactor safety is the best feature of the IPSRs. Large LOCAs and related mechanisms are eliminated by design because of the integral configuration, while small LOCAs are eliminated as safety concerns because of the small containment volume, which allows for a rather high back-pressure buildup upon a LOCA or depressurization event, so that an inventory of water sufficient to prevent core uncovery is always maintained, i.e., the vessel and containment are thermal-hydraulically coupled. Most of the LWR energy release accident scenarios (7 of the 8 Class IV accidents considered in AP600) are also eliminated.

In the area of reactivity control, the lack of soluble boron eliminates the possibility of boron dilution accidents, while the internal control-rod drive mechanisms prevent reactivity insertion accidents due to control rod ejection.

While the IPSRs are designed to prevent severe accidents, they would perform better than reference in the unlikely case that core damage did occur. In-vessel retention is guaranteed by cooling of the outer surface of the vessel by the water in the vessel cavity. A large water inventory in the containment, a robust containment, passive hydrogen control systems, and passive containment cooling system (PCCS) all ensure long and effective holdup of the fission products in the event of a serious accident. In particular, the large inventory of water in the containment will dissolve and retain many of the fission products.

From the standpoint of predicting the reactor behavior under normal and off-normal conditions, it is judged that all dominant phenomena in the IPSRs can be accurately characterized and studied experimentally (including full-scale setup) both in separate effect and integral experiments. Also, the IPSR designs feature on-line monitoring of all dominant parameters during operation: this is a needed, essential feature for an integral configuration.

The IPSR summary score in the area of reliability and safety is 0.43.

2.1.4 Concept Strengths and Weaknesses in Economics

While the IPSRs do not enjoy the economy of scale of large monolithic plants, they are greatly simplified, eliminating all primary structures outside the primary vessel (i.e., piping, pumps, pressurizer. and steam generator vessels) as well as many safety systems, such as the emergency core cooling system of the reference. Their modular size lends itself to an economy of multiples and shop fabrication. Similarly, a significant operations and maintenance (O&M) cost reduction is possible due to the projected four-year maintenance shutdown and even more infrequent refueling, but uncertainties do exist. The capital at risk will be obviously low and the construction time has the potential to be significantly reduced. Small, modular IPSRs can be deployed in series, such that the first units operating during construction of the subsequent ones generate a cash flow.

The IPSR ranks in the top group in Economics, due to the modular designs having reasonable overnight construction costs, relatively short construction durations, and relatively small capital at risk.

Because of their reliance on proven LWR technology, the IPSRs require a relatively limited amount of development. Development programs have been already conducted by CAREM and MRX; the SMART prototype will provide a wealth of information, while IRIS will utilize the applicable AP-600 technology (e.g., the passive containment cooling testing), plus ad hoc testing performed overseas.

The IPSR summary score in the economics area is 0.58.

2.2 Simplified Boiling Water Reactors

2.2.1 Concept Set Description

The BWR designs, successfully promoted by the GE Co 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 ABWR design—a Generation III plant. The Generation IV SBWR designs submitted for consideration are summarized in Table 2. 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 2. Summary of SBWR 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 (GE, 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 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 (NRC) 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:

1. These BWRs are all direct-cycle LWRs with conventional energy conversion systems and efficiencies (with the exception of the desalination plant, W22).

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

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

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

5. 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 UO_2 for very high burnup (Long Operating Cycle Simplified BWR, LSBWR), and MOX rods (ESBWR).

6. 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.

7. 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.

8. 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).

A quantitative assessment was developed by TWG1 for the ESBWR, which is discussed in the next three subsections. The ESBWR is a 4000-MWth BWR 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. Adequate natural circulation behavior has been achieved using shorter fuel and an improved steam separator (to reduce the pressure drop in the primary system), and a seven-meter chimney to enhance the driving head.

The ESBWR uses isolation condensers (ICs) for high-pressure inventory control and decay heat removal under isolated conditions. The IC 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 Core-Cooling System (GDCS), instead of relying on pumps and their associated support systems. The ESBWR uses an automatic depressurization system (ADS) to depressurize the vessel. Containment heat removal is provided by the PCCS, 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 ICs, are located in cooling pools external to the containment.

The quantitative assessment score sheet for the ESBWR and the overall rollup can be found the score sheet appendix on this CD.

2.2.2 Concept Strengths and Weaknesses in Sustainability

Standard LEU fuel was assumed in the scoring of the ESBWR concept. The U-235 enrichment is 4%, the average discharge burnup is 45 MWd/kgHM, and the thermal efficiency is 33%. Therefore, the ore utilization, mass of waste, volume of waste, long-term heat output, long-term radio-toxicity, environmental impact, separated materials, and spent fuel characteristics are identical to the reference ABWR (scored same as reference). Because the ESBWR makes extensive use of passive safety systems, the decay heat removal requires no actuation, and there are multiple means for core cooling, the sabotage resistance is judged to be somewhat better than the reference. In addition, improvements have been made in the plant design to separate the spent fuel storage from the reactor building. The overall sustainability score is 0.06, i.e. about the same as reference.

2.2.3 Concept Strengths and Weaknesses in Reliability and Safety

The main strengths of the ESBWR, compared to the reference ABWR, lie in the area of safety and reliability. Because this is a natural circulation reactor, the outage rate due to failure of the core-cooling pumps will be eliminated. In addition, the simplified design and use of advanced diagnostic systems will help minimize outage rates. However, outage rates are, in general, driven by external events, problems in the turbine building, and human error, which are not much effected by any Generation IV design, so the reliability is only scored slightly better than the reference. The simplified plant design and use of extensive diagnostics also has the potential to significantly reduce worker exposures and the opportunity for worker accidents.

The ESBWR has negative temperature and reactivity coefficients and a control-rod system that is similar to the ABWR. However, because it is a natural-circulation reactor, the reactivity cannot be controlled by varying the core flow rate with the re-circulation pumps. However, it is judged that, overall, the ESBWR will perform similarly to the reference in the area of reactivity control.

The criteria “reliable heat removal,” “dominant phenomena have low uncertainty,” “long fuel response time,” and “long system time constants” are all scored significantly better than the reference because of the ESBWR’s extensive use of passive systems and large inventory of water to ensure reliable removal of the decay heat. The ESBWR eliminates the high-pressure coolant injection system and instead relies on a depressurize-and-reflood strategy. The ESBWR has an ADS, a GDCS, and a PCCS in addition to the active residual-heat removal system. These systems are designed to prevent core uncovery for at least 3 days following any design basis accidents (DBAs).

The criteria “source term,” “mechanisms for energy release,” and “long and effective holdup” are also somewhat better than the reference because of the use of passive safety systems and a large water inventory in the ESBWR. Although the ESBWR is designed to prevent severe accidents, in the unlikely event of core damage the energy release mechanisms are reduced because in-vessel retention and cooling are assured with the GDCS. Also, retention of soluble fission products is enhanced by the large inventory of water in the containment.

The ESBWR summary score in the area of reliability and safety is 0.36.

2.2.4 Concept Strengths and Weaknesses in Economics

The ESBWR buildings and systems are reduced by about 20 to 25% compared to the ABWR on a per MWe basis. Use of existing infrastructure and components will reduce the capital cost uncertainty. This plant is relatively similar to the reference and its fuel cycle is identical to reference. So its operating cost should be similar to the reference, with potential for modest improvements due to the reduced number of active components, which may decrease the O&M costs somewhat. The ESBWR construction duration should be similar to the recent Japanese ABWRs because the plants are similar in size.

The central values for the overnight construction cost, construction duration, operating cost and bus bar cost of electricity of the ESBWR are $1200/kWe, 40 months, $15/MWh and $35/MWh, respectively.

The amount of ‘new’ development and engineering for the ESBWR is very limited. This reactor is almost fully developed.

The ESBWR summary score in the economics area is –0.01 because of the high capital at risk due to the large size of the plant, and because of the relatively high bus bar cost of electricity. In other words, despite the 20 to 25% reduction in systems and buildings claimed by GE, TWG1 scores this concept as about the same as the reference in economics.

2.3 Pressure Tube Reactors

2.3.1 Concept Set Description

Several advanced pressure tube reactor design concepts have been proposed as Generation IV reactors (see Table 3). 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.

Table 3. Generation IV pressure tube reactor concepts.

Concept	Key Features	Sponsor
W6, NG-CANDU	-Light-water coolant -Heavy-water moderator in calandria -Slightly enriched uranium (SEU) fuel -Significantly smaller calandria than CANDU-6 -Higher outlet temperature and plant efficiency than CANDU-6	AECL
W28, Passive Light-Water Pressure-Tube Reactor	-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 LWR	-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 NG-CANDU design is improved economics, achieved principally through a capital cost and construction schedule reduction. Key features that enable the improved economics are a reduction in the heavy water inventory, an increase in the outlet temperature and the plant thermal efficiency, a smaller core, and a design based on modular construction. The NG-CANDUs also have enhanced safety and enhanced fuel cycle flexibility. The Passive Pressure Tube Reactor design is focused on passive safety, whereas the High Conversion Pressure Tube Reactor design is focused on fuel cycle optimization.

The NG CANDU was selected as representative of this reactor design concept set and a quantitative assessment was prepared by TWG1 for that concept alone. NG CANDU is the latest in a series of CANDU reactors designed by AECL and it evolves from the CANDU-6 design, which has been sold around the world. The latest CANDU 6 design is the twin unit Qinshan project in China, which will come on line in 2003. NG-CANDU has adopted an evolutionary approach, accommodating significant changes to design while retaining traditional CANDU strengths of:

- Modular horizontal fuel channel
- Available simple, economical fuel bundle design
- Separate cool, low-pressure heavy water moderator with back-up heat sink capability
- On-line/at-power fuelling
- Fuel cycle flexibility with high neutron efficiency

- Passive moderator/shield tank heat sinks surrounding the pressure tube core
- Two robust, quick acting, passive shutdown systems

The following key development steps or improvements are incorporated into the design concept for the NG-CANDU:

- Replacement of heavy water in the reactor coolant system with light water coolant
- SEU oxide fuel at increased burnup in CANFLEX fuel bundles
- More compact core design with reduced lattice pitch, reducing heavy water inventory and giving highly stable core neutron flux
- Higher coolant system and steam pressure and temperature
- Small, negative reactivity coefficients
- Enhanced passive safety systems
- Higher thermal efficiency
- More compact design with ease of construction and localization
- Configured as a twin 650-MW plant.

The result of these features is a plant that is inexpensive, low risk, and reliable, with a short construction schedule and several safety enhancements, including passive safety and sabotage protection. The plant is available in the near term with a once through SEU cycle, but more sustainable fuel cycles are under development.

The quantitative assessment score sheet for the NG-CANDU and the overall rollup can be found in the score sheet appendix on this CD.

2.3.2 Concept Strengths and Weaknesses in Sustainability

The evaluation is based on the use of LEU fuel with 3% U-235 enrichment, a core average discharge burnup of 46 MWd/kgHM, and a plant thermal efficiency of 35%. The 3% U-235 and 46 MWd/kgHM fuel is a higher burnup fuel than the CANDU-6 fuel (natural uranium and 7.5 MWd/kgHM) or the current NG-CANDU design (2% U-235 and 20 MWd/kgHM), but is realizable within the Generation IV time frame.

With this fuel form and parameters the uranium utilization, long-term heat output, environmental impact, and proliferation characteristics of the NG-CANDU are similar to reference. It is judged that the environmental impact advantages arising from elimination of boron for reactivity control purposes are offset by the tritium generation from activation of the heavy water moderator. The mass of waste is somewhat worse (greater than) reference and the volume of waste is somewhat less than (better than) reference. Also, the long-term radiotoxicity is slightly better than reference and the sabotage resistance of the NG-CANDU is somewhat improved by the use of passive systems.

Overall, this reactor is judged to provide a small improvement over current LWRs in the area of sustainability. The sustainability summary score for the NG-CANDU is 0.08.

2.3.3 Concept Strengths and Weaknesses in Reliability and Safety

The NG-CANDU is being designed with input from reliability-centered maintenance, probabilistic safety analysis (PSA), and feedback from the experience of operating CANDU plants. Because of the evolutionary nature of the design, most of this information can be used directly to improve the design and reduce the likelihood of failures and outages. Design features that were found to cause difficulties in the CANDU-6 plants have been addressed in the NG-CANDU design. Because the reliability of the CANDU-6 plants is similar to the LWR reference, the reliability of the NG-CANDU will be better than reference.

Traditional CANDUs exhibit worker exposure rates similar to ALWRs, despite the presence of tritium from activation of the heavy-water coolant and moderator. The NG-CANDU concept eliminates the heavy-water coolant, and thus offers potential for some improvement. Also, automation and remote sensing for routine system reliability tests and inspection, and component and material refinements in key systems such as fuel handling are targeted to reduce dose by more than 50%.

Unlike traditional CANDUs all reactivity coefficients are negative and small for the NG-CANDU, minimizing uncontrolled reactivity insertion under both overheating and overcooling transients. These reactivity coefficients are stable throughout the fuel lifetime in the core, unlike LWRs where the reactivity and core management can change considerably, particularly with the use of very-high burnup fuels. Moreover, the NG-CANDU has three independent safety shutdown systems, located in the low-temperature low-pressure moderator pool, which greatly increases their reliability, and simplifies inspection and maintenance. Finally, because of online refueling the excess reactivity in the core can be maintained low enough to eliminate the possibility of prompt criticality, and the need for soluble boron in the coolant can be eliminated. Therefore, it is judged that the NG-CANDU will be much better than reference in the area of reactivity control.

The NG-CANDU is rated better than reference on reliable heat removal because the system can thermo-siphon to remove decay power on loss of AC power. The NG-CANDU can also switch to a shutdown cooling system at full temperature and pressure. Also, there is a large inventory of heavy water in the moderator pool.

The long-term thermal response of the NG-CANDU design during accidents is better than that of the ALWR because of the presence of the separate, low-temperature inventories of water in the moderator and the shield tank surrounding the core. This means that the response time of the system to some severe accident sequences and the rate of core degradation are slower. However, fuel damage can occur upon loss of flow in a pressure tube. During severe accidents, core degradation is slowed by the large inventory of water in the core, while having the safety control rods in the moderator tank prevents molten-fuel re-criticality. The calandria tank will retain most fission products upon core damage. Also, the nature of the fuel channel core design makes it possible to perform full-scale tests of key phenomena associated with channel safety behavior, but some safety codes must be validated against tests in well-scaled facilities. The heat removal system located in the moderator tank operates continuously.

Long and effective holdup is also scored better than reference because the NG will have high integrity containment for all internal and external events. The NG-CANDU design will have a redundant isolation system to ensure containment integrity in the event of a large release event. (Under Canadian licensing rules, the design must be analyzed and shown to be safe for events with small release coupled with an assumed failure of the ventilation isolation.) In addition, the NG-CANDU plant has a very large surface area of piping (in the feeders and headers) that can act to holdup and mitigate fission product material release from the core. Also, the calandria/shield tank assembly will act to prevent or delay the introduction of large quantities of molten core debris into the containment where it can produce aerosol fission products. However, the CANDU plants do not have 100% passive containment cooling.

The overall NG-CANDU summary score in the area of reliability and safety is 0.35.

2.3.4 Concept Strengths and Weaknesses in Economics

The NG-CANDU is being designed with the goal of significantly improving the economics of existing reactors. The heavy-water coolant and all related systems have been eliminated. The coolant is now light water. Due to the higher moderating power of light water and to the increased U-235 enrichment, the number and size of the pressure tubes have been drastically reduced. This resulted in a much more compact core, moderator tank, shield tank, and containment building. The number of steam generators and the number of flow splits of the low pressure turbine have been halved, which enabled a substantial downsizing of the turbine building. The volume of piping, valves, fittings and cabling has been reduced by 40% compared with the CANDU-6 system. Finally, the construction schedule has been shortened to 36 months through the aggressive use of pre-fabricated components, modularization, and 3D CAD design of the construction processes.

All of the above have resulted in a very competitive nuclear system with overnight construction cost, operating cost and bus bar cost of electricity of about $1000/kWe, $11/MWh, and $30/MWh, respectively. Note that the capital and operating cost estimate for the NG-CANDU plant have a substantial degree of credibility because AECL has real procurement, construction and operation experience. AECL has built several CANDU-6 reactors over the past decade in different countries and is in the process of completing two units for Qinshan Phase III in China. The NG-CANDU cost estimate is based on the mature and up-to-date CANDU equipment cost database with supplier input. Because the NG-CANDU is an evolution from the CANDU 6 design, the cost knowledge is directly applicable.

The amount of *new* development and engineering for NG-CANDU is limited. In addition, the design conditions have been selected to be modest extensions of the existing R&D database for key CANDU components and materials. The NG-CANDU design includes improvements in the safety and operating margins in key areas to reduce the need for expensive qualification and validation testing.

The overall NG-CANDU summary score in the economics area is 0.66.

2.4 Supercritical Water-Cooled Reactors (Fast and Thermal Spectrum)

2.4.1 Concept Set Description

SCWRs are a class of high temperature, high pressure water-cooled reactors that operate above the thermodynamic critical point of water (374°C, 22.1 MPa or 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 have also been proposed. Cylindrical as well as spherical fuel elements (i.e., pebble bed) are also being currently considered. The key advantages to the concept that are derived from the use of higher temperatures during heat addition include:

- Significant increases in thermal efficiency can be achieved relative to current generation LWRs. Estimated efficiencies for SCWRs are in the range of 44-45% compared to 32-34% for state-of-the-art LWRs.
- The higher enthalpy content of the SCW results in a much lower coolant mass flow rate per unit core thermal power. This leads to (a) a reduction in the reactor coolant pumping power and (b) reduced frictional losses in the steam lines due to lower steam mass flow rates.
- A lower coolant mass inventory results from the reduced coolant density as well as lower reactor coolant system heat content. This results in lower containment loadings during a design basis LOCA and the possibility of designing small containment buildings.

- No boiling crisis (i.e., departure from nucleate boiling or dry out) exists during normal operation 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 smooth heat transfer deterioration in SCWRs and a boiling crisis will occur if the primary coolant system is depressurized during an accident.
- Because the coolant does not undergo a change of phase, the need for steam dryers, steam separators, re-circulation and jet pumps, as well as steam generators, is eliminated.

It is important to point out that the SCWR is more akin to a gas cooled rather than a light water reactor. The primary system pressure is about 3 times the pressure in a BWR and it operates with a much lower coolant density at a much higher exit temperature than in a BWR.

Six supercritical concepts were submitted for consideration, including one concept that has four variants (the SCW-cooled CANDU: W6). The concepts are summarized in Table 4.

Table 4. Proposed Generation IV SCWR concepts.

Concept/ Organization	Concept Name	Moderator	Rating MWe	Outlet Temp, °C	Net Efficiency %	Comments
W21, Univ. of Tokyo	Thermal spectrum SCWRs	H_2O	1700	508	44	Once-through, direct cycle
TWG1	Fast spectrum SCWRs	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- SCWR feeds very high pressure turbine. Very high pressure turbine exhaust feeds steam generator with traditional indirect cycle
W6-4, (Super-critical CANDU)/ AECL	CANDU-ALX2	D_2O	1143	625	45	Dual-cycle- SCWR feeds a very-high-pressure turbine. Very high-pressure turbine exhaust feeds steam generator and core inlet regeneration.
W2, (Pebble Fuel)/ Pacific Northwest National Laboratory (PNNL), USA	Pebble bed BWR w/Super-critical Steam	H_2O	200	540	40	Fluidized bed of SiC-PyC-coated UO_2 particles in supercritical steam

The Japanese supercritical light water reactor (SCLWR) with a thermal spectrum has probably been the subject of the most development work in the last 10 to 15 years. The SCLWR reactor vessel is somewhat similar in design to an ABWR. High-pressure (25.0 MPa) 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 about 510°C and delivered to a power conversion 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 direct cycle SCWR can also be designed to operate as a fast reactor. The primary difference between a thermal and fast SCWR is in the amount of moderator material in the core region. The fast spectrum reactors do not need additional moderator material, whereas, the thermal spectrum reactors need significant moderator material in the core. The Japanese thermal spectrum SCLWR uses water rods for neutron moderation, however, other direct-cycle designs have been developed with solid moderator material in the core region.

These two reactor concepts, the SCLWR and the fast spectrum version of the SCLWR, were chosen for complete quantitative evaluation during the TWG1 deliberations. Two variants of the thermal design were also qualitatively discussed; i.e., a heavy water moderated thermal reactor within CANDU pressure tubes and a thermal reactor with small (2 to 5 mm) spherical fuel pebbles with TRISO coatings. In order to properly assess these two reactor concepts, a series of key assumptions were made about both reactor concepts:

1. The nominal reactor outlet temperature was assumed to be 510ºC with an operating pressure of 25MPa. These were chosen to be consistent with the base design suggested by Japanese and European researchers. Similar nominal values have been considered by the heavy water moderated design as well as the pebble bed design.
2. The fuel cycle is quite versatile in this reactor design. For our evaluation of the thermal spectrum reactor we assumed a LEU fuel cycle with 5% U-235 and a burnup of 50 MWd/kgHM. For the evaluation of the fast spectrum reactor, it was assumed that the fuel cycle was multi-recycle of plutonium and minor actinides using proliferation resistant advanced aqueous reprocessing of the spent fuel.
3. Because of materials compatibility issues it was assumed that the fuel cladding would likely not be Zircaloy, but a more high-temperature, corrosion-resistant metal such as stainless steel or a high chrome, high nickel steel.
4. The passive safety improvements of the SBWR were considered to be quite compatible with the current SCLWR design and assumed to be part of it. These passive features include the improved ADS, the reactor Isolation ICs, as well as the GDCS and the PCCS.

Quantitative assessment score sheets for both the thermal and fast spectrum SCWRs and the overall rollups can be found in the score sheet appendix on this CD.

2.4.2 Concept Strengths and Weaknesses in Sustainability

The evaluation for the thermal SCWRs is based on an LEU fuel cycle with 5% U-235 enrichment (that is about 1% higher than an equivalent LWR fuel because of the use of stainless steel or nickel alloy cladding), a thermal efficiency of 45%, and a burnup of 50 MWd/kg_{HM}. The higher thermal efficiency affords some improvement of the sustainability indices based on a per unit electric energy generated (i.e., about 1/3 less mass and volume of spent fuel, long-term heat output, and radio-toxicity) with the exception of the uranium utilization, for which the thermal efficiency effect is offset by the somewhat higher enrichment requirements. The proliferation-resistance characteristics (fuel form and plutonium composition at discharge) are similar to reference.

Overall, the thermal spectrum SCWR are judged to provide a modest improvement over current LWRs in the area of sustainability. The sustainability summary score for the thermal SCWRs is 0.13.

The evaluation for the fast spectrum SCWRs is based on the use of a MOX and minor actinide multi-recycle. The core average discharge burnup is 80 MWd/kg_{HM}, the plant thermal efficiency is 45%, and the heavy metal loss per recycle pass is less than 1%. The spent fuel reprocessing technology is proliferation-resistant advanced aqueous reprocessing, similar to that used for the liquid-metal-cooled oxide-fueled reactors. These reactors share all the sustainability advantages typical of fast reactors with multiple recycling, i.e., a uranium utilization two orders of magnitude better than the once-through fuel cycle, a drastic reduction of mining requirements, minimal generation of long-lived waste and heat. The minimization of mining (only partially offset by the waste generated by reprocessing), the lower generation of activated corrosion products and the lower inventory of waste to dispose of also result in better environmental impact of these systems.

The spent fuel reprocessing technology will be advanced aqueous, which will assure that the fissile material is protected by an intense radiation barrier. Moreover, the high burnup and the presence of the minor actinides will make the spent fuel very unattractive for weapon proliferation.

The overall sustainability summary score for the fast SCWRs is 0.62.

2.4.3 Concept Strengths and Weaknesses in Reliability and Safety

The reliability of the SCWRs is uncertain. The reduction in piping, components, valves and needed operator actions gives a chance for improvement. By contrast, the very high reactor primary coolant system pressures, temperatures, and corrosive environment are sources of concern. The SCWRs could have materials corrosion or stress-corrosion cracking problems. The fast-spectrum version of the SCWR will also be subjected to significant fuel-cladding and core internals radiation damage (75 to 200 dpa).

A unique characteristic of the SCWRs is the reduced inventory of activated corrosion products because the coolant is not re-circulated in the core (like in PWRs and BWRs), which, combined with the simplified plant design and use of extensive diagnostics, has the potential to significantly reduce worker exposures. However, the separators and dryers in BWRs make it possible to retain in the water non-gaseous fission products and fuel particles released from any failed fuel rods. Without that filter, the SCWR steam turbine may get more contaminated in the unlikely event of a fuel failure.

These reactors can be designed with passive safety systems similar to the ESBWR. Core uncovery can hopefully be eliminated by means of a depressurize-and-re-flood strategy based on an ADS, a GDCS and a PCCS in addition to the active residual-heat removal system. However, the direct-cycle SCWRs have a much lower reactor vessel coolant inventory and it is not clear that passive safety systems will work effectively in SCWRs. Also, there are issues associated with the nuclear-thermal-hydraulic stability of SCWRs that have not been resolved. As discussed in Chapter 3, viability research is needed in these areas.

The overall SCWR summary score in the area of reliability and safety is 0.22 and 0.19 for the thermal and fast designs, respectively. The difference is mainly due to the more troublesome reactivity control for the fast core.

2.4.4 Concept Strengths and Weaknesses in Economics

Superior economics is the main thrust for the SCWRs. The cost savings occur because (1) the thermal efficiency is increased from about 34 to about 44 or 45%, (2) the SCWR systems eliminate the need for many major expensive components designed to handle coolant boiling in traditional LWRs, i.e., the steam generators and the pressurizer in PWRs, and the steam separators, steam dryers, re-circulation and jet pumps in BWRs, and (3) the overall plant is significantly reduced in size. For example, for the same thermal output, the reactor pressure vessel weight is reduced from 910 to 750 tons (18%) and the containment volume is reduced from 17,000 to 7,900 m^3 (54%). Moreover, due to the high enthalpy content of SCW, the main coolant pumps are significantly reduced in size and rating, the number of steam lines is halved, and the number of low-pressure turbines and condensers drops from 3 to 2. However, some equipment cost increases may occur due to the higher temperature and pressure requirements. It is judged that at a minimum this design will have an overnight-construction-cost reduction proportional to the increased thermal efficiency, with potential for substantial further improvement due to plant simplification. The increased thermal efficiency will also result in a significant reduction of the O&M costs because of the much larger electric power produced with a somewhat smaller plant and approximately the same staff size for O&M as the reference.

The central values for the overnight construction cost, construction time, operating cost, and bus bar cost of electricity of the thermal SCWR are $900/kWe, 50 months, $10/MWh and $25.4/MWh, respectively, while for the fast SCWR are $900/kWe, 50 months, $13/MWh and $28.4/MWh, respectively. The difference in the operating cost is primarily due to reprocessing.

SCWRs will require significant fuel cladding and core structural materials development and testing as well as fuel bundle testing, including loop testing in existing test reactors. It will also require significant separate effects and scaled integral thermal-hydraulic safety testing. The balance of plant materials and equipment will be the same as currently used in the SCW-cooled fossil fired plants and will not need much development. However, a small demonstration plant will be needed to fully demonstrate the concept before full sized plants are built. The total development costs for the SCWRs are estimated to exceed $1,000M.

The overall summary score in the economics area is 0.51 for the thermal SCWR and 0.42 for the fast SCWR.

2.5 High Conversion Water-Cooled Reactors

2.5.1 Concept Set Description

The high conversion water-cooled-reactor concepts are essentially typical LWRs, but with a tight triangular pitch fuel rod lattice to minimize moderation and produce a fast spectrum essential to achieve the high conversion ratio. Most do this within a BWR plant design but two designs are based on the PWR. Since the high HCBWR 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 relate 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 use MOX fuel with recycle of the fissile material, including the minor actinides. In general, the spent fuel reprocessing technology proposed is either proliferation resistant advanced aqueous or dry (AIROX type) reprocessing.

The features of the various high conversion core designs are summarized in Table 5.

Columns 1 and 2 of Table 5 list the acronym used and the principal designer. There are more variations in this concept set, but these represent the ones documented for TWG1. The third column in the table gives the reactor type, i.e., the nuclear steam supply system used. In general it is the ABWR design that would be used; however, one concept has integrated their core with a more advanced version, the ABWR-II, and one intends to use aspects of the SBWR to improve safety. The 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; natural circulation is used.

Table 5. High conversion water-cooled-reactor core designs.

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

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 of the 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 5.

To evaluate this concept set, individual evaluations were solicited from each of the concept originators, and several were received. However, it was felt that they were not all consistent and that it would be a better idea to submit one evaluation that TWG1 felt comfortable with. It was decided not to consider the PWR concepts because of concerns about the potential costs of using heavy water in a PWR system. For much the same reason the Hitachi SSBWR design, with heavy water being diluted with light water during each fuel cycle, was not considered. Among the BWR concepts, it was felt that:

1. The SBWR approach should be eliminated because not all their safety features would be appropriate for a high conversion design with its very tight lattice. Also, there was concern about whether a natural circulation approach would even work with such a tight lattice design.

2. The ABWR-II provides significant advantages over the ABWRs in safety and reliability and economics. Six Japanese BWR utilities and three BWR plant venders, GE Company, Hitachi Ltd., and Toshiba Corporation, have jointly been developing the ABWR-II for about a decade. A reference 1700 MWe plant design has been developed which will achieve approximately 15% lower power generation costs and have only one-tenth the risk of core damage, compared to the ABWR.

Therefore, TWG1 decided that the HCABWR-II concept submitted by Hitachi was our preferred concept to assess. The quantitative assessment score sheet for the HCABWR-II and the overall rollup can be found in the score sheet appendix on this CD.

2.5.2 Concept Strengths and Weaknesses in Sustainability

The evaluation for the HCABWR-II is based on the use of MOX fuel and minor actinide multi-recycle. The core average discharge burnup is 45 to 60 MWd/kgHM (the design limit on burnup in these reactors is uncertain), the plant thermal efficiency is 34%, and the heavy metal loss per recycle pass is less than 1%. The spent fuel reprocessing technology will be either proliferation resistant advanced aqueous reprocessing, similar to that used for the liquid-metal-cooled oxide-fueled reactors, or dry reprocessing. These reactors share most the sustainability advantages typical of fast reactors with multiple recycling, i.e., a uranium utilization two orders of magnitude better than the once-through fuel cycle, a drastic reduction of mining requirements, minimal generation of long-lived waste and heat. (Note that the current scoring by the Fuel Cycle Crosscut Group does not give this concept full credit for the improvements in “long-term heat output” and “long-term radio-toxicity” that is given to the other Generation IV fast reactors, and is probably incorrect.) The minimization of mining (only partially offset by the waste generated by reprocessing) and the lower inventory of waste to dispose of also result in a somewhat better environmental impact of these systems.

The spent fuel reprocessing technology will be advanced aqueous or dry reprocessing, which will assure that the fissile material is protected by an intense radiation barrier and makes the scoring for “separated materials” and “spent fuel characteristics” the same or slightly better than reference. Moreover, the burnup and the presence of the minor actinides will make the spent fuel very unattractive for weapon proliferation.

The overall sustainability summary score for the HCABWR-II is 0.59, significantly better than the thermal spectrum reactors, but lower than the other fast reactors.

2.5.3 Concept Strengths and Weaknesses in Reliability and Safety

The reliability and worker safety (both routine exposures and accidents) of the HCABWR-II is judged to be about the same as reference. The design improvements of the ABWR-II and use of advanced diagnostic systems will help minimize outage rates and exposures. But the additional radiation damage in the core region could cause some reliability problems. Also, the fuel cycle facilities may add a small amount of worker exposure and risk. In general, the ABWR-II is a relatively small evolution from the current ABWRs and therefore should score similar to reference for the SR1 criterion.

In the area of reactivity control, the HCABWR-II design has negative reactivity coefficients and a control system similar to the reference ABWRs. Therefore, it is scored similar to reference.

In the area of accident behavior, the ABWR-II makes use of passive and active safety systems. The ABWR-II design includes both the PCCS developed for the SBWRs and a passive core heat removal system that removes heat from the primary cooling system through heat exchangers, which are independent of the PCCS. The PCCS and passive core heat removal system provide backup to the active ECCS for severe accident scenarios such as an extended station blackout or failure of the active cooling systems and provide alternative ultimate heat sinks to seawater. PSA evaluations have shown that these passive systems have reduced the core damage frequency for internal events by about one order of magnitude. Therefore, the “reliable heat removal,” “long fuel response time,” “long system time constants,” and “long and effective holdup” are scored somewhat better than reference. However, very little analysis has been made available to TWG1 to score these criteria, and the scoring for these criteria must be considered uncertain. The “source term” and “mechanism for energy release” from the ABWR-II during an accident will be essentially the same as reference because the plant, including the active safety systems, is so similar to the reference ABWR.

The overall reliability and safety summary score for the HCABWR-II is 0.21, or only slightly better than reference.

2.5.4 Concept Strengths and Weaknesses in Economics

The design goal of the ABWR-II is to achieve a 30% reduction in power generation costs compared to the reference. Part of that cost reduction is obtained by pushing the capacity up to 1500 MWe from 1350 MWe. The design changes also include fewer, but larger size fuel bundles and valves. It is not clear to TWG1 that the designers can achieve a 30% cost reduction compared to the reference, so the overnight capital costs are judged to be only about 10% better than the reference.

In general, the ABWR-II is a relatively small evolution from the current ABWRs and, therefore, the production costs should be relatively similar to (maybe 10% less than) the reference. The MOX recycle fuel will cost somewhat more that the reference LEU fuel, however, the design improvements of the ABWR-II and use of advanced diagnostic systems will help minimize the O&M costs.

The ABWR-II should have a 40-month construction duration based on current Japanese practice with the ABWRs. Regarding the capital at risk, the ABWR-II will be a 1500 MWe plant with a 40-month construction period and an overnight capital cost of about $1350/kW. This works out to about $2,025 million. The profitability of the ABWR-II was calculated using the EMG equations assuming a capacity factor of 92%. The result is a profitability greater than $37/MW-hr, which is worse than reference.

Little development cost is needed for systems external to the core since they are same as the ABWR plants. However, new cladding materials need to be developed ($100M), and neutronics and thermo-hydraulics methods and verification experiments are needed ($200M). In addition, the fuel cycle (advanced aqueous or dry reprocessing and MOX fuel fabrication) needs further development ($500M). A demonstration plant is not needed.

The overall economic summary score for the HCABWR-II is –0.11, or slightly worse than reference.

3. TECHNOLOGY GAPS AND REQUIRED R&D

The current state of the knowledge, gaps between what we know and what we need to know, and the proposed R&D programs for both thermal and fast spectrum SCWRs, IPSRs, pressure tube reactors (primarily the NG-CANDU), and the HCABWR-II are discussed in Sections 3.1 through 3.4 below.

3.1 Supercritical Water Cooled Reactors

The important research needs are summarized in Section 3.1.1. The research plan for the fuel rod cladding, core internals, and other materials work is presented in Section 3.1.2. The research plan for the safety work is presented in Section 3.1.3. The research plans for power-flow stability and plant-design follow in Sections 3.1.4 and 3.1.5. Note that there is no research discussed on the fuel cycle because the Fuel Cycle Crosscut Group will address that subject in more depth. In general, the thermal spectrum SCWRs will use conventional LEU fuel. The fuel itself is fully developed; however, new cladding materials and fuel bundle designs will be needed. (The designs for the thermal spectrum SCWR will need significant additional moderator i.e. water rods or solid moderation, the designs for the fast spectrum SCWRs will require a tight pitch, but high neutron leakage to create a negative void coefficient.) The fast spectrum SCWR will use mixed plutonium-uranium oxide fuel with advanced aqueous reprocessing (technologies being developed for a number of fast reactor concepts). Again, new cladding materials will be needed.

The proposed R&D program discussed below is detailed and involves a large number of issues. The material in Sections 3.1.2 and 3.1.3 (materials and safety) primarily describes needed “viability” research. The material in Section 3.1.4 (instability) primarily describes needed “performance” research. The material in Section 3.1.5 describes the work needed to optimize the plant design.

3.1.1 Summary of the Research Needs for the SCWR Reactor Concepts

The research needs for both the SCW-cooled thermal and fast spectrum reactors are summarized below. The research needs for the pressure tube and pebble bed version of the SCWRs are listed separately below.

- Identify, or develop where necessary, materials and alloys for the fuel rod cladding, core internals, and balance of plant that:
  - Can resist corrosion and stress corrosion cracking (SCC) in SCW at operating temperatures.
  - Have acceptable dimensional and microstructural stability. This implies an understanding of irradiation-induced changes such as growth, swelling, helium bubble formation, dislocation microstructure, precipitate microstructure, and irradiation induced composition changes.
  - Have acceptable strength, embrittlement, and creep resistance.
- Understand the effects of radiolysis on the coolant water chemistry and on the corrosion and SCC of materials in SCW systems. Optimize the water chemistry, including the development of appropriate hydrogen and noble metal water chemistry.
- Reduce the uncertainties in the SCW thermal-hydraulic transport properties.
- Further development of appropriate fuel cladding to coolant heat transfer correlations for SCWRs under a range of fuel rod geometries.
- Obtain SCW critical flow measurements, and develop appropriate models and correlations.

- Measure the integral LOCA thermal-hydraulic phenomena in SCWRs and validate appropriate computer models.
- Measure and model the fuel rod cladding ballooning during LOCAs.
- Perform SCWR design optimization studies.
- Perform key safety analyses and associated experiments to qualify and quantify the reactor safety systems needed to achieve specific safety and economics goals. This would include investigations to understand factors that influence the reliability of the passive systems, and establishment of reliability models for the passive systems.
- Perform experiments and analyses to (a) understand power-flow instabilities in SCWRs (including side-to-side instabilities), (b) identify the important variables affecting these phenomena, and (c) generate maps that identify stable operating conditions. Density-wave instabilities, coupled thermal-hydraulic-neutronic instabilities, and natural circulation instabilities are all possible and need to be assessed.
- Understand and model the transport of corrosion and fission products from the reactor system to the turbine island.
- Perform reactor and plant design optimization studies in the following areas: fuel assemblies, CRDMs, internals, reactor pressure vessel, pressure relief systems, coolant clean-up systems, coolant chemistry control systems, power control logic, turbine, re-heaters, de-aerators, start-up systems and procedures, in-core sensors, plant parameter control systems, reactor/containment building, etc.

Research needs specific to the pebble bed concept:

- Perform experiments and model the behavior of the pebble bed fuel particles including corrosion, erosion, cracking, or other forms of degradation expected during operation.
- Determine and model the coolability of packed-bed geometries under accident conditions.
- Perform experiments and develop models of the thermal-hydraulic behavior in a pebble-bed fuel reactor geometry during normal, off-normal, and accident conditions.
- Determine how to measure the burnup of the pebble bed fuel particles to determine fuel discard/re-injection during refueling.
- Determine how to best fabricate large TRISO particles.

Research needs specific to the pressure tube SCWRs:

- Develop pressure tubes capable of carry SCW pressures and temperatures.
- Develop thin wall collapsed cladding for SCW coolant conditions.

3.1.2 R&D Plan for SCWR Materials and Structures

At present, no single candidate alloy has been identified as the probable alloy for use as either cladding or structural materials in either the thermal or fast spectrum SCW reactors. The SCW environment is totally unique in terms of nuclear experience and little data exist on the behavior of materials in SCW under irradiation and in the temperature and pressure ranges defined by the reactor design envelope. As such, the R&D plan for the cladding and structural materials in the SCWRs will focus on acquiring data and a mechanistic understanding related to the following key property needs: corrosion and SCC, radiolysis and water chemistry, dimensional and micro structural stability, and strength and creep resistance.

The discussions below address the needs for both thermal and fast spectrum SCWRs. It should be noted that the fast SCWR design would result in greater doses to cladding and structural materials than in the thermal design by a factor of 5 or more. Instead of maximum doses for the core internals in the 10-30-dpa range in the thermal design, the maximum doses for the core internals in the fast reactor design could reach 100–150 dpa. These higher doses will result in considerably greater demands on the structural materials in terms of the need for irradiation stability and effects of irradiation on embrittlement, creep, corrosion, and SCC. The generation of helium by transmutation is an important consideration because it can lead to embrittlement at high temperatures. While the He generation rate is a function of the neutron spectrum and the alloy composition, (Garner et al., 1998, 1999, 2001) the differences in neutron energy spectra between the thermal and fast reactors will result in a substantially higher He/dpa production rate in the thermal design vs. the fast design. As such, He embrittlement will be most important in thermal SCWRs.

For each set of materials properties, a goal stating the minimum information needs for implementation of this reactor concept will be stated. The next section will focus on the current state of knowledge of the material behavior or property. Then, the gap between what is known and what needs to be known is established from the previous two sections. Finally, the needed research program including schedules, costs, and facilities is discussed. The identified gaps and the needed research are primarily "viability" research, however, the later portion of each research project does provide some "performance" information.

The special needs for several additional versions of the SCWR design including the pressure tube heavy water moderated design and the pebble bed design are discussed last.

3.1.2.1 Corrosion and Stress Corrosion Cracking

Goal

The structural materials used in SCW reactor systems must be able to maintain integrity over a 60-year lifetime. In terms of corrosion and SCC, this means that corrosion rates must be such that the total metal consumption will not compromise the strength or performance of the component or the components must be designed for easy and low cost replacement. Stress corrosion crack initiation and propagation must likewise, be confined to the extent that the components will last for the 60-year lifetime. The fuel cladding must resist corrosion and SCC for about 5 years of core residence time in the thermal spectrum versions of the SCWRs and for up to 10 years in the fast spectrum versions. Adequate corrosion resistance in subsequent spent fuel storage is also necessary.

Current State of Knowledge

Critical reactor systems and components for operation in SCW. Using the materials of construction of current generation LWRs as a starting point and superimposing the likely conditions of a SCWR operating as a fast reactor, it is clear that major changes need to occur in materials selection. The proposed design conditions for a SCWR are compatible with the operation of ceramic UO_2 or mixed UO_2-PuO_2 fuel and few changes are foreseen in the fuel currently being used or developed. However, the Zircaloy or zirconium-niobium alloy cladding currently used in LWRs will be unacceptable in SCW conditions. The corrosion rate of Zircaloy cladding in PWR water is acceptable at the current burnup limits and continued alloy developments have led to some extended lifetime for related alloys. However, the corrosion rate of Zircaloy increases significantly in steam and at higher temperatures. The combination of increased oxidation rate in a highly oxidizing environment and higher temperature make Zircaloy a doubtful cladding material for this reactor type. As such, alternate fuel cladding materials must be identified.

Conventional boiler/steam turbine fossil-fired power plants have been operating with superheated steam for a number of years and can be used as a starting point for the development of materials for the SCWR concept (Viswanathan, et al. 2000). Steam-temperatures as high as 600°C are being used and research is being conducted to push the operating temperature up by another 50–100°C over the next 30 years. These plants have developed 9–12% Cr ferritic steels for use up to 540°C by increasing their strength, creep resistance and fatigue resistance. Depending upon the aggressiveness of the environment, oxidation and waterside corrosion become problems between the mid 500°C range and 600°C and alloying additions are being investigated in order the push the usability of these steels into the 600°C range. Above 600°C, austenitic iron- or nickel-base alloys will be required to achieve the needed corrosion resistance, but these alloys will need to be strengthened to increase creep resistance and creep-rupture life.

The bolts, springs, fasteners, etc., that comprise the core internal components in LWRs are typically made from austenitic stainless steels (304, 316), single phase solid solution nickel-base alloys (600, 690), or precipitation hardened nickel base alloys (718, X-750, 625). Their overall corrosion performance has been acceptable, but numerous problems have occurred, and continue to occur, with SCC of these components. The problem is enabled by the water environment and aggravated by radiation and temperature. While this class of alloys is generally attractive for SCWR application, significant testing and development will be required to identify the most promising alloys and verify their suitability.

The reactor pressure vessel in LWRs is made from low alloy, ferritic steels and clad with stainless steel to protect it from the water environment. It is likely that the stainless steel clad will adequately protect the vessel from the SCW environment as well, minimizing the concern over a corrosion issue. The principal concern with reactor pressure vessel steels is irradiation hardening and embrittlement, leading to reduced fracture toughness. However, the vessel temperature in a thermal spectrum SCW reactor is expected to be about the same or only slightly higher (30–50°C) than in an LWR, which would lead to a reduction in embrittlement for the same dose. The temperatures would be similar in the fast spectrum SCWR; however, the thermal shield and water gap will need to be designed to control the vessel dpa to current limits. Hence, the reactor pressure vessel is not expected to be a limiting component in terms of degradation in a SCW environment. However, advanced steels could be used to increase the design lifetime of reactor pressure vessels compared to current reactors.

The last component of concern is the turbine. The turbine materials will see much higher temperatures and pressures than are used in current LWRs. However, materials have been developed to operate in superheated steam at higher pressures in fossil plants and this experience will aid in the selection of materials. While turbine materials will ultimately need to be addressed, it is not deemed to be one of the critical, design limiting components that needs to be addressed in the early stages of SCW reactor development.

Critical materials issues in SCW. The existing database on the corrosion and SCC of alloys in SCW is sparse. Besides SCC in SCW, accelerated corrosion and intergranular stress corrosion cracking (IGSCC) may also occur in the pre-heat and cool-down sections of the circuit. Results suggest that there are two critical regimes – the SCW itself and the pre-heat and cool-down locations of the reactor. The prime problem in both regimes appears to be SCC. While the corrosion rate increases with temperature, the critical failure mode shifts from general corrosion (wastage) to SCC in both regimes. Experiments by Latanision (1995) showed that severe IGSCC occurred in Hastelloy C-276 at intermediate temperatures where the solubility of inorganic salts is low. This could present a situation similar to the secondary side of a LWR steam generator. The solubilities on the secondary side of LWR steam generators are such that significant precipitation of solids out of solution results in the formation of films that can trap impurities and raise the local concentration levels to the point where localized corrosion, wastage, and IGSCC become severe. As such, the intermediate temperature regime (between the primary water temperature in current designs and SCW temperatures) also needs to be studied to determine whether this may present an aggressive environment.

From a mechanistic point of view, it seems particularly interesting to study the effect of the transition regime (from subcritical to SCW) on the SCC behavior of the structural materials. Kriksunov and Macdonald (1995) have suggested that in aerated water, this transition from subcritical to supercritical conditions was accompanied by a change in the corrosion mechanisms from those of a liquid phase, i.e. ionic mechanisms (coupled cathodic and anodic reactions), to those of a gas phase (oxidation). Recent constant extension rate tension (CERT) tests performed on Alloy 718 (precipitation hardened nickel base super-alloy) both in subcritical and supercritical aerated water confirmed this suggestion (Founier et al. 2001). In aerated subcritical water, smooth specimens strained at 10^{-6} s^{-1} did not exhibit SCC. In contrast, smooth specimens strained in aerated SCW showed an important loss of ductility, coupled with an intergranular fracture mode. This difference in behavior was attributed to the difference in the crack initiation conditions, via the difference in corrosion mechanisms, which is consistent with the fact that SCC of Alloy 718 in hot water is related to the existence of an initial defect. Crack initiation in aerated SCW was found to result from oxidation and swelling of the niobium primary carbides. It is therefore possible that oxidation controls both the initiation and propagation stage of stress corrosion cracks of nickel base super-alloys in aerated SCW. The observation that Alloy 690 (30% Cr) is not sensitive to SCC in aerated SCW is consistent with this hypothesis (Fournier et al. 2001).

Fuel cladding. Candidate alloys for which the resistance to corrosion and SCC is relatively well known and for which we also have a fair understanding of the irradiation behavior fall into 2 classes: (1) the austenitic iron-base stainless steels (e.g., 304, 316 and more advanced alloys) and the high chrome austenitic alloys (e.g., Alloys 800 and 690), and (2) corrosion resistant ferritic alloys such as HT-9 and more advanced ferritic/martensitic alloys such as modified 9Cr steels. Some of these alloys have undergone extensive testing in high dose, high dose rate, and high temperature environments that extend into those similar to fast reactor core conditions. The maximum doses (100–150 dpa) and temperatures (450–500°C) are similar to a fast reactor environment. The principal effects of a high temperature, high dose environment that must be withstood by a fuel cladding are irradiation induced hardening, irradiation induced creep, void swelling, and phase stability. Many data are available from the fast reactor program on austenitic stainless steels in this environment. There is also a good body of literature on the behavior of HT-9 and more advanced ferritic/martensitic steels in similar environments. Corrosion and SCC data of irradiated materials are limited (Klueh and Harries 2001) at these temperatures, but there is a growing database on the behavior of austenitic alloys under LWR conditions due to the recent research on the irradiation assisted stress corrosion cracking (IASCC) problem.

The thermal and radiation stability of these alloy classes are promising. However, the added demands of supercrticial water containing a high oxygen level contribute a high degree of uncertainty as to the suitability of these alloy systems. The selection of the most promising candidate materials could be made based on available data on the radiation stability (hardening, creep, swelling, phase stability), and on extrapolations of the corrosion and SCC behavior in subcritical water and steam.

Core internal components. For internal component application, initial attention should be focused on the two alloy classes discussed for fuel cladding with the addition of precipitation hardened nickel-base Alloys 718 and 625 (although it must be recognized that the super alloys will likely be severely embrittled at temperatures above 540°C for doses above 10 to 20 dpa, e.g., see Ward et al. 1976). Alloy X-750 is also in this class but its SCC performance in LWR applications has been poor, so it should not be considered. From available data, the solid solution strengthened austenitic alloys (304, 316 and more advanced alloys) are an attractive choice for this application. They exhibit excellent general corrosion resistance in aqueous environments above 300°C and are generally phase stable under irradiation at this temperature (although void swelling is a concern). However, they also suffer from

IGSCC in this temperature range and the degree of degradation is accelerated by both temperature and irradiation. IGSCC that is accelerated or induced by irradiation is termed IASCC and is a significant and generic problem in the austenitic stainless steel components in reactor cores the world over. The problem is exacerbated by temperature, radiation damage, high corrosion potential, and a cold worked microstructure. While this alloy class remains as the top prospect for SCW reactor core applications, little is known about IGSCC under the relevant conditions. At temperatures of 600°C and above, the possible sensitization by thermal aging of the austenitic stainless steels (e.g., type 304) must also be taken into account (intergranular corrosion).

A second class of austenitic, solid solution alloys that should be considered includes the high chrome iron-based Alloy 800 and nickel-based Alloy 690. Nickel-based Alloy 600 has suffered from a history of IGSCC problems in PWR steam generators and control rod drive feed-throughs and is being phased out of service in LWRs. It is being replaced by Alloy 690 or 800. Both of these alloys contain higher chromium content at the expense of the nickel and laboratory tests in oxidizing and reducing conditions and in primary and secondary water have shown that both are highly resistant (though not immune) to IGSCC. Service experience thus far has been good (there was one stress corrosion crack reported in a German Alloy 800 tube). Limited SCC tests have been conducted on Alloy 690 in SCW at 400°C and 25 MPa and have shown no indication of IGSCC and comparable ductility to tests in air at the same temperature (Fournier et al. 2001). Hence, these alloys represent promising candidates for core component applications. (The Alloy 690 does have a different coefficient of thermal expansion than the iron-based alloys.) A major concern for the nickel-base alloys is radiation embrittlement due to grain boundary precipitation and helium bubble formation at temperatures above ~500-550°C (Ward et al. 1976, Vaidyanathan et al. 1982, Mills 1992). There are no known neutron irradiation data for Alloy 690.

The third class of alloys, ferritic HT-9 and more advanced ferritic/martensitic alloys containing 9–14%Cr, are also promising from both a corrosion and radiation stability standpoint. All three classes should be evaluated for their radiation stability and their known corrosion and SCC experience in high temperature water and steam.

Gaps between What We Know and What We Need to Know

- We don't know the corrosion behavior of any of the candidate materials [austenitic iron- or nickel-base alloys or ferritic, ferritic-martensitic or ferritic-oxide-dispersion strengthened (ODS) alloys] in pure SCW, outside of the experience in the fossil plants. We need to know corrosion rates and corrosion mechanisms for each of the candidate materials.
- We need to know the effect of the SCW temperature in the range 280-620°C on the corrosion behavior of each of the candidate materials. Note that the entire reactor pressure vessel is operating at supercritical pressures, however, the inlet temperature for the direct cycle plants is expected to be ~280°C. At some point partly up the core the coolant reaches the pseudo critical temperature and the coolant changes from being somewhat liquid to more of a gas. So the relevant temperature range begins at 280 and extends up to about 620°C. (There may also be indirect cycle SCWR designs with coolant always above the pseudo critical temperature.)
- We need to know the effects of oxygen, hydrogen, and impurities, $SO_4^{=}$, Cl^-, etc. on the corrosion behavior of each of the candidate materials.
- We need to know the compound effects of irradiation damage and radiolysis of the water on the corrosion behavior of each of the candidate materials.
- We need to know the SCC susceptibility of all the candidate alloys in pure SCW, including both the susceptibility to crack initiation and the crack growth behavior.

Proposed R&D Program

The SCWR corrosion and SCC research program should focus on obtaining the following information:

- Corrosion rates of candidate alloys in SCW at temperatures between 280 and 620°C. The corrosion should be measured under a wide range of oxygen and hydrogen contents to reflect the extremes in dissolved gasses, and also in long-term experiments at 620°C.
- Composition and structure of the corrosion films as a function of temperature and dissolved gasses.
- The effects of irradiation on the corrosion as a function of dose, temperature, and water chemistry. Does irradiation accelerate the oxidation process?
- SCC as a function of temperature, dissolved gasses, and water chemistry.
- The effects of irradiation on SCC as a function of dose, temperature, and water chemistry. Does irradiation accelerate the oxidation process?

The corrosion and SCC R&D program will be organized into three parts: an extensive series of out-of-pile corrosion and SCC experiments on un-irradiated alloys, companion out-of-pile corrosion and SCC experiments on irradiated alloys, and in-pile loop corrosion and SCC tests. It is envisioned that at least two, and maybe as many as four, out-of-pile test loops would be built, some addressing the corrosion issues and others addressing the SCC issues. At least two such loops should be built inside a hot cell in order to study pre-irradiated material. Facilities to pre-irradiate samples prior to corrosion and SCC testing will be required. Doses of 10–30 dpa will be required to support the thermal design and doses into the 100-150-dpa range will be required to support the fast reactor version of the design. This work should be carried out over a 6- to 10-year time span for unirradiated materials and the same for irradiated materials

Accelerators capable of producing high currents of light ions may be utilized to study irradiation effects on corrosion and SCC in a postirradiation mode at substantially lower cost than reactor irradiations. The results of experiments with austenitic stainless steels over the past 10 years have shown that irradiation with protons in the 3-MeV range causes changes to the microstructure (dislocation microstructure, void microstructure, segregation, and precipitation), hardness, and SCC susceptibility that closely emulate those from neutron irradiation (Was et al. 1999, 2002). The advantage of using accelerators for conducting irradiation studies is the shorter irradiation time compared to reactor irradiation (by 100 to 1000 times) and the reduced sample activation. Both factors lead to a greatly reduced cost and more rapid acquisition of data. Cost estimates would be $1 million/yr for accelerator-based irradiations that would serve to screen alloys, conditions, microstructures, etc. to reduce the scope of in-reactor irradiations and to provide guidance on the key alloys/conditions on which to focus.

About mid-way through the out-of-pile work (years 3–5), at least one, and more likely two, in-pile test loops should start operating under both fast and thermal spectrum irradiation conditions (a total of 3 to 4 loops). The in-pile loops will be used to study corrosion, SCC, and water chemistry control issues (see Section 3.1.2.2 below). We will probably need about 10 years of in-pile testing in these loops to obtain all the required data to support both the viability and performance phases of the development of the thermal spectrum version of the SCWR and maybe as much as 15 years of testing to obtain the needed information for the fast spectrum SCWR. The Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL) is a likely location for the thermal spectrum testing. The fast spectrum testing will need to be done outside the U. S., probably in Russia or Japan. A postirradiation characterization and analysis program will accompany reactor- and accelerator-based irradiations beginning in year 5 and extending for a 10-year period.

Costs for this program would be on the order of $4 million per year for the out-of-pile testing of the un-irradiated material (four loops at various laboratories and universities), and $4 million per year for out-of-pile testing of irradiated alloys in hot cell facilities (two loops at laboratories and universities). In-pile testing will require about $6 million per year per loop, and about $3 million per year for the post-irradiation examinations and analyses. Assuming 5 years of out-of-pile testing on un-irradiated alloys, 5 years of out-of-pile testing on irradiated alloys, and 12 years of in-pile testing with two or three loops, the total will be about $250 to 350 million. This cost should support the development of both the thermal and fast spectrum SCWR.

3.1.2.2 Radiolysis and Water Chemistry

Goal

Understand the effect of radiolysis on water chemistry and on the corrosion and SCC of materials in SCW systems.

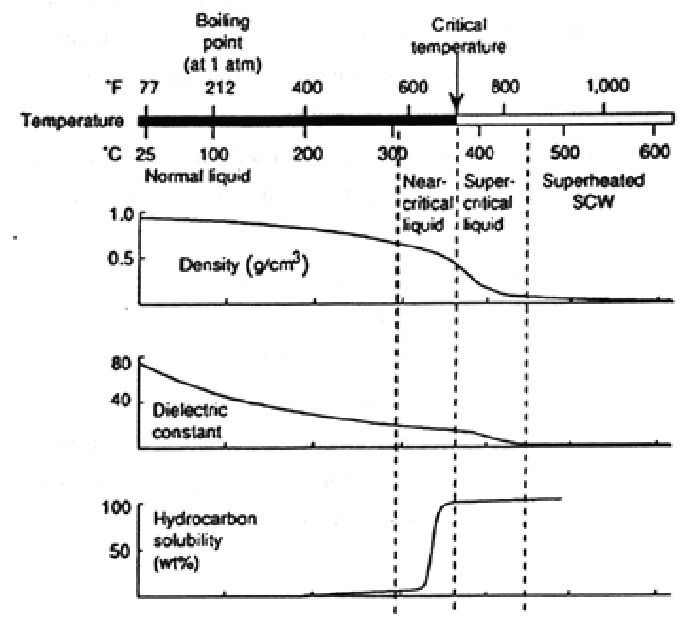

Figure 1. Properties of water in the supercritical region for the pressure range 218-300 atm. Taken from Latanision 1995.

Current State of Knowledge

Water has unique properties above its critical point (374°C and 221 atm for pure water). Under these conditions, water is a fluid with properties intermediate between those of a liquid and those of a gas. Figure 1, taken from Latanision (1995), illustrates some of the important properties of SCW. The density of water changes dramatically from about 1 g/cm^3 in the liquid phase to <0.001 g/cm^3. Between 374 and 400°C, the density of water is between 0.5 and 0.1 g/cm^3. The dielectric constant of water changes from 80 at room temperature to 5 to 10 in the near critical range and to 1 to 2 at 450°C. In addition, the ionization constant (K_w) for water changes from 10^{-14} to 10^{-23} at supercritical conditions. Also, hydrogen bonding is virtually extinguished at supercritical conditions (Franck 1976). As a consequence, its solvent properties more closely resemble those of a low-polarity organic than room temperature water (Tester et al. 1993).

Ultimately we need to know the corrosion potential for the water/metal interface system(s) in the core of the SCW reactor. This will be enormously influenced by the water radiolysis, just as in lower temperature reactors. Water will be decomposed to give molecular H_2, H_2O_2, and O_2 as stable products, with solvated electrons, H atoms, OH radicals, and hydronium ions as reactive intermediates. In order to predict the corrosion potential, we need to know the chemical (redox) potential of all the species present. This requires that we know the radiolytic yields of the intermediates and their recombination rates, all as a function of density and temperature, as current research has shown tremendous changes in the chemistry over the range of physical properties to be encountered in the SCWR. The good news is that thanks to lower solubility of ions and corrosion products in SCW, there is the possibility that corrosion may be less severe than in current PWRs.

We also need to understand the effectiveness of hydrogen and noble metal water chemistry in suppressing the electrochemical potential of the SCW.

Gaps between What We Know and What We Need to Know

- Chemical potential and recombination rates of H_2, H_2O_2, O_2, and various radicals in SCW over a range of temperatures (280-620°C) and fluid densities.
- Effect of radiation (type—neutrons, gammas, as well as flux) on radiolytic yields as a function of temperature and fluid density.
- Formation of other species (e.g., nitrogen oxides, carbonate, metal corrosion products) by radiolysis.
- Effectiveness of hydrogen and noble metal water chemistry on suppressing the electrochemical corrosion potential of the SCW.

Proposed R&D Program

The SCWR water chemistry research program should focus on obtaining the following information:

- The complete radiolysis mechanism in SCW as a function of temperature and fluid density, including the effects of radiation on radiolysis yields
- The chemical potential of H_2, H_2O_2, O_2, and various radicals in SCW over a range of temperatures (280–620°C)
- Recombination rates of various radicals, H_2, H_2O_2, and O_2 in SCW over a range of temperatures (280–620°C)
- Formation and reaction of other species by radiolytic processes
- Effectiveness of hydrogen and noble metal water chemistry on suppressing the electrochemical corrosion potential of the SCW.

Two general avenues of research are envisioned to obtain this information. First, beam ports and accelerators can be used to irradiate SCW solutions and study the characteristics of the recombination processes in some detail. This information will be integrated into a model of the water radiolysis mechanism. Second, water chemistry control studies can be performed using the in-pile test loops needed for the corrosion and SCC research discussed above. The additional water chemistry studies can be performed for about $10 million per year over a period of about 8 years, for a total cost of $80 million. (The cost of the in-pile and out-of-pile loops is in the estimate for the corrosion and SCC experiments above; this is the incremental cost of the water chemistry studies.)

3.1.2.3 Dimensional and Microstructural Stability

Goal

Establish that irradiation-induced changes to the cladding and structural materials due to growth, swelling, helium bubble formation, dislocation microstructure, precipitate microstructure and radiation-induced composition changes are known for the systems and conditions for use in SCW environments and that these changes will not compromise the structural integrity of the clad and structural components for the design life of the reactor plant. This research is particularly important for the fast spectrum version of the SCWR, but is also needed for the thermal spectrum version.

Current State of Knowledge

The behavior of irradiated structural materials in the 280–350°C range is reasonably well known, but not always well understood. In austenitic iron- and nickel-base alloys at or below 300°C, swelling is not significant due to the difficulty of nucleating and growing voids (Zinkle et al. 1993). Above 300°C, void nucleation depends sensitively on the alloy composition, solute additions and He production rate. Below 350°C, irradiation-induced precipitation is not a significant problem (Zinkle et al. 1993, Maziasz and

McHargue 1989). It can, however, occur in locations where the solubility limit for a particular solute is reached due to radiation induced segregation (RIS). RIS-induced precipitation at grain boundaries may result in an undesirable brittle phase. RIS occurs throughout this temperature range and can cause significant composition variations at strong sinks such as grain boundaries. While segregation of the major alloying elements Fe, Cr, Ni is fairly well characterized and can be reasonably well modeled, the understanding of the behavior of minor elements that are believed to migrate as interstitials is poor and modeling capabilities are inadequate. The dislocation microstructure is also reasonably well characterized (Zinkle et al. 1993, Maziasz and McHargue 1989, Garner 1994). Outstanding issues involve the process of loop nucleation and the loop character.

Ferritic/martensitic alloys offer the potential for better microstructure stability while providing good strength with moderate corrosion resistance. These alloys are more resistant to swelling because of the long incubation period for void nucleation. However, there is only limited data or understanding of other potentially important processes such as RIS and dislocation microstructure development under irradiation (Klueh and Harries 2001).

In the temperature range 350–620°C, candidate materials will require higher strength and greater resistance to diffusion-driven processes such as RIS and void formation and growth. Helium diffusion and precipitation under irradiation will also become more important. In this temperature range, the austenitic alloys pass through their peak in swelling. At the upper end of the range, void swelling is expected to be minimal. The use of solute additions (e.g., Ti-stabilized stainless steel) as well as oversized solutes may promote recombination and delay the onset of swelling so that doses of as much as 100 dpa may be reached before swelling becomes too large to accommodate by design. Helium generated from thermal neutron capture in Ni will migrate to grain boundaries and result in grain boundary bubble embrittlement at temperatures above ~550°C. Here again, solute additions and fine-scale precipitation may be important in trapping He at vacancy-solute or precipitate clusters to delay the aggregation of He into bubbles. RIS and irradiation-induced precipitation becomes an increasingly important issue. RIS will peak in the 400–500°C range, while near the upper end of the range the high concentration of thermal vacancies will suppress RIS. The dislocation loop density will decrease sharply and the loop size will coarsen throughout the temperature range. At the upper end of the range (>500°C), the microstructure should resemble an annealed condition with very few loops and a low network dislocation density (Zinkle et al. 1993). Irradiation-induced precipitation will become increasingly important with temperature in this range. The existing database in this temperature regime is geared toward higher displacement rates typical of fast reactor conditions, and at slightly lower maximum temperatures than those expected in the SCWR outlet. The majority of the available results are also reflective of the high fast-to-thermal flux (dpa/He) ratio characteristic of fast reactors, although a substantial database is available from mixed (thermal) spectrum reactors.

In this temperature regime, all microstructure features change quickly with temperature. Little is understood about the complex interactions that could occur between microstructure features such as dislocations, voids/bubbles, RIS, precipitates when each is a very sensitive function of temperature. The interplay between these features and their relative sensitivities to temperature will be important to understand for this alloy system to be applied in this temperature regime.

Dispersed oxide precipitates or other second phase particles will be required for austenitic alloys to maintain adequate strength at the upper end of the temperature range (>550°C). The behavior of the precipitates under irradiation and their dose/temperature evolution is largely unknown, as is RIS, void swelling, and dislocation microstructure interaction with the precipitates. The impact of an evolving precipitate structure on the dislocation microstructure, void nucleation and growth, and RIS at the interfaces are critical challenges. Very little data are available on these oxide dispersion strengthened alloys and fabrication and joining techniques could well become a limiting factor.

Ferritic/martensitic alloys provide the potential to achieve doses above 200 dpa due to their inherent resistance to swelling (Klueh and Harries 2001, Gelles 1996). Here again, oxide dispersion strengthening may be required at the upper end of the temperature range and the evolution of RIS, dislocation microstructure and precipitate evolution will be need to be better understood. There are only limited experimental data on RIS in these systems (Klueh and Harries 2001).

Solid solution Ni-base alloys provide adequate strength and creep resistance up through the intermediate portion of the temperature range and precipitation-hardened alloys are well suited to the upper end of the range. However, high He production and precipitation–induced grain boundary embrittlement will likely limit application in this temperature range to low (<20 dpa) dose applications (Ward et al. 1976, Vaidyanathan et al. 1982, Mills 1992). Nevertheless, precipitate stability and RIS in this system is not well understood, even in this low dose range.

Gaps between What We Know and What We Need to Know

- There are insufficient data on any alloy (austenitic, ferritic-martensitic) in a thermal reactor spectrum at high temperature where (n, α) transmutation reaction rates will be higher and will generate more He per dpa. Void nucleation and growth, the role of transmutation He in void stabilization and growth and grain boundary He embrittlement in the higher half of the temperature range needs to be measured.
- Further information is needed about the dislocation microstructure and (more importantly) on RIS in irradiated ferritic-martensitic steels over the entire temperature range, but especially at the upper end of the range. The same is true for austenitic iron- and nickel-base alloys at the high end of the temperature range
- Irradiation-induced precipitation and the behavior of precipitation-hardened alloys such as ferritic-ODS alloys and their joints (e.g., the heat effected zone due to welding) are poorly understood under SCWR irradiation and temperature conditions.

Proposed R&D Program

The SCWR dimensional and microstructural research program should be focused on obtaining the following information:

- Void nucleation and growth and the effect of He production on void stability and growth and He bubble nucleation and growth as a function of dose (up to ~30 dpa in a thermal spectrum and up to 150 dpa in a fast spectrum) over the temperature range 280–620°C.
- Development of the dislocation microstructure, precipitate microstructure and radiation-induced segregation as a function of dose (up to ~30 dpa in a thermal spectrum and up to 150 dpa in a fast spectrum) over the temperature range 280–620°C. The stability of oxide particles in irradiated ODS alloys would be included in this task.
- Knowledge of growth or irradiation-induced distortion as a function of dose (up to ~30 dpa in a thermal spectrum and up to 150 dpa in a fast spectrum) over the temperature range 280–620°C.
- Knowledge of irradiation-induced stress relaxation as function of tension, stress, material, and dose (see the requirements for bolts and fasteners).

While many of the test specimens for this work will be irradiated in the corrosion and SCC in-pile loops discussed above, accelerator-based irradiation offers a rapid and low cost alternative to the handling and analysis of neutron–irradiated material. Much of the needed information will be obtained during post-irradiation examinations over the 15-year period of the corrosion and SCC tests. In addition, some stand-alone capsule irradiation tests in test reactors should be performed in order to obtain scooping data on a range of candidate materials in a timely manner. The additional post-irradiation examination and analyses costs are expected to be about $3 million per year for a total of $45 million for neutron irradiated samples.

3.1.2.4 Strength, Embrittlement, and Creep Resistance

Goal

Strength must be maintained at the high temperatures expected in SCW reactor designs and time-dependent deformation must be kept below levels where dimensional changes can be accommodated by design. For example, the 10^5 h creep rupture strength at 600°C has been used by the fossil power plant industry as a benchmark for evaluating alloys for SCW service (Viswanathan et al. 2001). Further, reductions in temperature during outages must not result in embrittlement of internal components or fuel cladding over the life of the components.

Current State of Knowledge

Thermal and irradiation creep can be very severe in this temperature range for conventional austenitic stainless steels such as AISI-316 or AISI-304, and limits their use under high stresses to <600ºC. The irradiation creep rate tends to be a factor of two lower for ferritic and ferritic-martensitic steels (Garner 1994). Recently, a class of ferritic and ferritic-martensitic steels has been developed in which a very fine (~1 to 4 nm diameter) dispersion of oxide particles has been produced (Ukai et al. 1998, Klueh et al. 2000). The best of these ODS alloys have been shown to maintain this fine dispersion even under thermal creep conditions of elevated temperature and stress. If this behavior is maintained under irradiation, the ODS steels may increase the upper temperature limit of the ferritic-martensitic steels by 100 to 200°C, and the operating stress limit in the 350 to 600°C temperature range. Limited irradiation data on a French ferritic ODS steel up to 600°C indicates that Chi phase formation can lead to crack nucleation at low plastic strains. This same alloy also showed evidence of oxide particle dissolution after irradiation to 80 dpa at ~500°C. Conversely, oxide particle dissolution was not observed in MA957 oxide dispersion strengthened ferritic steel after irradiation up to 200 dpa (Gelles 1996).

Both solid solution and precipitate-strengthened nickel-base alloys have also been investigated in this temperature range. Many of these alloys are limited by softening and creep under irradiation above 500°C, and by the formation of brittle intergranular second phases. The high nickel content leads to the formation of high levels of helium from nuclear transmutation reactions initiated by thermal neutrons. Even relatively modest amounts of helium can significantly reduce ductility in these materials and may accelerate fatigue crack growth. Some nickel-based super-alloys may be more resistant to intergranular embrittlement and creep deformation, but will likewise be limited by the formation of helium.

Systems used in a load-following mode will require more attention be paid to fatigue and creep-fatigue interactions in this temperature regime. The required analysis methodology will depend on the cyclic loading frequency, absolute stress level, and temperature. The potential effect of RIS, second phase formation and exposure to the reactor coolant must also be considered. Thus, the analysis will be design and material specific. A reasonable database exists only for the austenitic stainless steels, and to a lesser extent, some of the advanced ferritic and ferritic-martensitic steels. Irradiation data are lacking on other potential alloy systems such as the ODS steels and high nickel alloys.

Additional concerns for ferritic and ferritic-martensitic steels are the same as for pressure vessel steels, i.e., radiation or thermal aging effects on toughness and the ductile-to-brittle-transition-temperature (DBTT). Similar radiation-induced material embrittlement issues and research/ development needs exist for each of the primary alloy choices for high-dose cladding and core components.

Gaps between What We Know and What We Need to Know

- Creep and irradiation creep behavior of austenitic steels up to 620°C (for advanced alloys; a substantial database already exists for 304 and 316 stainless steel).
- Creep of ferritic, ferritic-martensitic steels and ODS steels, including welded joints, up to 620°C.
- Effect of irradiation on creep on ferritic, ferritic-martensitic steels and ODS steels (a substantial database exists for several 9- and 12-Cr ferritic/martensitic steels).
- Fatigue, fatigue crack growth rate, and creep-fatigue interaction especially at the high end of the temperature range.
- Effects of RIS and second phase formation on fatigue in austenitic, ferritic, ferritic-martensitic and ODS alloys at temperatures up to 620°C.
- Fracture toughness behavior for many candidate ferritic and ferritic-martensitic alloys is not well known for alloys in both the irradiated and unirradiated conditions. The fracture toughness behavior is also largely unknown in austenitic alloys for doses above 10 dpa.
- Linkage between radiation-induced microstructure changes and embrittlement or fracture toughness is largely unknown for many of the most promising, high temperature ferritic alloys.

Proposed R&D Program

The SCWR strength, embrittlement and creep resistance research program should be focused on obtaining the following information:

- Tensile properties (yield strength, ultimate tensile strength, elongation, reduction in area) as a function of dose over the range 10–30 dpa (thermal design) and 100–150 dpa (fast design) and temperature over the range 280–620°C
- Creep rates (primary and secondary) in candidate alloys in the dose range 10–30 dpa (thermal design) and 100–150 dpa (fast design) and temperature range (280–620°C) and as a function of applied stress
- Creep and creep rupture mechanisms for the same dose, temperature and stress conditions as used for creep rate measurements
- Creep-fatigue interactions and dependence on cyclic loading frequency, baseline versus load-following, effects of RIS on creep-fatigue
- Fatigue crack growth rate data in irradiated materials at 280–620°C
- Time-dependence of plasticity and high temperature plasticity
- Microstructural impact of creep-fatigue and feed back loop
- Helium embrittlement at operating temperatures (slow strain rate testing)
- Fracture toughness as a function of temperature and DBTT for each of the candidate alloys in the unirradiated condition (quasi-static and dynamic strain rates)
- Fracture toughness as a function of irradiation temperature (280°C to 620°C) and dose (to 30 dpa for thermal spectrum and to 150 dpa for fast spectrum)
- DBTT and helium embrittlement as a function of dose and irradiation temperature.
- Interaction between radiation-induced aging and fracture toughness/DBTT

- Changes in microstructure and mechanical properties following DBAs. Given the transient performance regimes, structural materials must be able to withstand accident conditions without compromising their mechanical property integrity (creep resistance, strength, embrittlement).

The research program will be aimed at high temperature performance of both irradiated and un-irradiated alloys and also at low temperature performance of irradiated alloys. High-temperature testing will include yield property determination, time dependent (creep) experiments and also the effect of fatigue loading with a high mean stress. This program will be conducted first on un-irradiated alloys over a period of 8 years at a level of about \$3 million per year. Midway through the program, testing will begin on irradiated materials for a period of 10 years at a level of \$5 million per year. The low temperature fracture toughness/DBTT program will require 10 years of effort at \$3 million per year. The total program will be funded at about \$100 million.

3.1.2.5 Special Considerations for Heavy Water Moderated Pressure Tube Type SCWRs

AECL is investigating various SCW-cooled versions of their NG-CANDU. The NG-CANDU is a light water cooled, heavy water moderated pressure tube reactor similar to the current CANDU-6 design, except that the core region is much more compact, the fuel is slightly enriched, and it operates at slightly higher core outlet temperatures and, therefore, plant efficiency. Converting the NG-CANDU design into a SCWR will require considerable materials development for both the pressure tubes and the fuel cladding. The pressure tubes in the SCWR CANDU will be required to carry both the very high-temperature and high-pressure load of the SCW (the SCWR direct cycle pressure vessels only see the higher pressure because the inlet coolant is brought in along the pressure vessel wall). This will require special insulation on the inside surface, a considerably thicker pressure tube wall, and outside cooling (AECL is investigating such a design). Even so, it is not clear that the zirconium based niobium alloy pressure tube material currently used in CANDU plants can be made to work under such conditions. Also, the thin wall, collapsed fuel cladding currently used in the CANDU plants will probably not withstand such high temperatures, and a new cladding material may be needed.

3.1.2.6 Special Considerations for Pebble Bed SCWR

It is claimed that the silicon carbide outer coating fabricated by chemical vapor deposition (CVD) can provide protection in air, water, and steam indefinitely at temperatures of 450 to maybe 600ºC and provide protection for a few days at temperatures as high as 1600ºC. One referenced investigation (Hurtado et al. 1992) exposed small fuel elements to temperatures between 600 and 1400ºC for 24 hours. In another investigation (Filippov and Bogiavlensky), the small fuel elements were exposed in a high-pressure water environment, i.e., (350ºC and 190 bar pressure) for 18 months. Conclusions from these investigations seem to be based upon negligible mass loss under the various test conditions.

Reaction rates of silicon carbide in air, steam and water are indeed extremely low at low and intermediate temperatures due to the formation of SiO_2 layers in both oxygen and water containing environments. The reactions associated with the formation of this phase are shown in Equations (1) and (2).

$$SiC + 3/2\ O_2 = SiO_2 + CO \tag{1}$$

$$SiC + 3\ H_2O = SiO_2 + 3\ H_2 + CO \tag{2}$$

In both cases, there would be an initial mass gain due to the pickup of oxygen exceeding the carbon loss. Due to these low oxidation rates, investigations to quantify and characterize the oxidation behavior of CVD SiC have been performed at higher temperatures, such as above 1200ºC. Opila and her associates (Opila 1994 and 1995, Opila and Hann 1997) have performed a series of experiments on CVD silicon carbide in dry oxygen, wet oxygen, and water vapor, respectively. Opila's (1994) tests in wet oxygen were performed between 1200 and 1400°C with a water vapor pressure of 0.1 atm. The author found that oxidation rates obtained from quartz chambers were only slightly higher than those obtained from dry oxygen. However, there was an approximate 10-fold increase in oxidation rates when alumina tubes were used with the moist environment. Increased transport of sodium and aluminum from the alumina tube to the specimen due to volatile hydroxides and the formation of less protective sodium alumino-silicate scales were given as the causes for the higher oxidation rates. A marked change in oxidation kinetics was found when the CVD SiC was tested in flowing 50% H_2O/50% O_2 between 1200 and 1400°C (Opila and Hann 1997). Linear rates of weight loss were observed after initial 20- to 30-hour periods showing parabolic weight gain. There are several possible volatilization processes that could be responsible for the weight loss. The authors proposed that $SiO(OH)_4(g)$ was the major contributor based upon mass spectroscopy evidence. This volatile product is expected to be the dominant contributor at the relatively low temperatures of interest for pebble bed SCWRs.

The authors reported that the variation of oxide thickness along a given specimen and weight change to oxide thickness correlation were poorer compared to those observed in dry oxygen. The nature of the oxide scale changed with temperature. Specimens exposed at 1200°C were amorphous. Scanning electron microscopy showed that some bubbles had formed in these scales. Exposures at 1300 and 1400°C produced crystalline scales of cristobalite. These scales contained cracks that the authors attributed to the β-to-α transformation that occurs near 250°C during cooling. Despite these differences, similar oxidation rates occurred between 1200 and 1400°C. The linear volatilization rate ranged between 2×10^{-3} and 5.5×10^{-3} mg SiC/cm^2·h. This would correspond to about 1.2×10^{-5} mm/h, or 1.2×10^{-2} μm/h.

Another recent study (More et al. 2000) by other investigators reported accelerated oxidation rates of CVD SiC when exposed in higher water pressure of 10 atmospheres at 1200°C. These authors observed the formation of a thick porous silica scale that formed above a dense silica layer at the scale-to-SiC interface. The dense layer at the interface appeared to reach and remain about 4 to 6 μm thick for exposures extending up to 4000 hours. The authors state that cracking, voids, scale spallation, and possible vapor re-deposition affects the precision of scale thickness measurements. However, they reported that 40 to 50 μm losses were likely from the SiC during 500-hour exposures. This would be a rate of about 1×10^{-1} μm/h or about 10 times higher that those observed by Opila and Hann (1997). The authors suggest that the porosity could be the result of CO and/or other product gases from impurities. The latter suggestion agrees with that proposed by Schiroky (1987) who suggested that CO or SiO could be responsible for bubbles, pits, and holes formed in CVD SiC when tested at higher temperatures, e.g., 1600 to 1800°C in air.

Initial results indicate that direct losses of material from the outer layer of the SiC coated pebbles should not be significant based upon the above rates. Firstly, there is a significant difference between the proposed operating temperatures of 280 to 500°C and the temperatures for the above experimental data. Even during the accident conditions analyzed by Tsiklauri et al. (2001) the temperatures would only approach 900°C for short times. This makes projected losses from the most severe 1200°C data, i.e., that from More, et al. (2000), appear quite small compared to the 80 μm thick layer of SiC on the small SFE. For example, the data indicate that only 1 μm would be lost during a 10-hour exposure at 1200°C.

However, the data above shows that there are several factors that influence the oxidation rates of SiC in water-containing environments. These include the water content, water or steam pressure, and impurities that may be derived from containment materials or inherent to the manufacturing process of the SiC. A better understanding of the influences of these parameters is needed. There are other forms of failure or deterioration that need to be considered. If exposed to the higher temperatures, the study by More et al. (2000) showed the formation of an outer porous, cracked layer. This material could become removed and suspended, or transported, in the coolant as the upward and cross flows rub pebbles together. These simultaneous mechanical interactions of the pebbles and oxidation mechanisms could also cause enhanced attack. Another consequence to a temporary excursion to around 1200°C could be the conversion to the silica layer to a crystalline form. This would cause the material to be susceptible to cracking when cooled through the β-to-α transformation temperature. In fact, factors that would cause the outer SiC or pyrolytic layers to crack and fail are the more likely safety concerns rather than failure by uniform attack or oxidation. Factors that might cause these are the inherent manufacturing quality of the SFE, thermal stresses, stresses and ductility changes resulting from irradiation, pressurization from fission gases, and mechanical interactions. Failure of the outer coating would then expose the inner material more susceptible to chemical interaction, primarily the porous pyrolytic carbon.

Additional issues to be considered for SCW-PBR:

- Radiation stability of pyrolytic carbon and SiC.
- Behavior of fuel particle collisions; erosion, cracking or other forms of degradation.
- Effect of high pressure on corrosion and SCC of particles.
- Fabricability of large TRISO particles/pellets.

3.1.2.7 Existing Facilities and Expertise

A brief overview of the facilities in participating countries is presented in this section.

Canada. The National Research Universal (NRU) facility is a 200-MWth reactor located at Chalk River Laboratories in Ontario Canada. NRU's large irradiation space has been an important factor in the testing of fuel bundles and fuel channel components for CANDU reactors. NRU can be outfitted to contain in-pile SCW loops to support SCWR R&D activities in the following areas:

- Testing of possible coupled thermal-hydraulic/neutronic instabilities, and
- Fuel and material irradiations.

Also, the ZED-2 research reactor can be for corrosion, as well as heat transfer, studies.

AECL also has a small-scale static autoclave to study the corrosion rates of core materials and out-of-core materials under supercritical conditions. The facility operates at up to 30 MPa and 500ºC. Recently eight materials currently operating in CANDU plants were studied for a 389-hour exposure in the static autoclave to neutral, deoxygenated SCW at 450ºC and greater than 25.3 MPa. The eight materials included steam generator alloys (Alloys 400, 600 and 800), zirconium alloys (Zircaloy-4, Zr 2.5 Nb), stainless steels (403 and 410), and carbon steel (A106B). This facility can be used to support SCWR R&D activities related to studying the corrosion behavior of prospective materials at supercritical conditions. Finally, the AECL has a high-efficiency channel that is a full-scale facility to study insulating materials for the CANDU X pressure tubes. The facility operates up to 30 MPa and 600ºC. This facility can be used to support SCWR R&D activities related to prospective insulating materials at supercritical conditions such as tests using reticulated (porous) and monolithic (non-porous) ZrO_2 material, and assessments of neutronic behavior and heat transfer under accident conditions.

United States. The ATR is the U.S. DOE's largest and most versatile test reactor, offering high thermal neutron flux and large test volumes for performing irradiation services. At maximum power, the unperturbed flux trap thermal flux is as high as 1.0×10^{15} (neutron/cm^2/sec.), and the fast flux is as high as 5.0×10^{14} (neutron/cm^2/sec.). The major test spaces in the ATR are the nine flux traps located in its core. A close integration of the flux traps and the driver fuel is achieved by the serpentine fuel arrangement shown in Figure 2. This fuel arrangement allows for closer alignment of the fuel on all sides of the flux trap in geometry not achievable in other standard rectangular or square test reactor configurations. Five of the flux trap positions are presently equipped with independent loops.

The pressurized water loop experiment is the most comprehensive type of ATR testing performed. A tube runs through the reactor core from vessel top to bottom and is attached to its own individual water system. The cooling system includes pumps, coolers, ion exchangers, and heaters to control test temperature, pressure and chemistry. A loop could easily be installed to specifically provide the necessary test capabilities needed to provide the parameters of a SCWR. For example, the existing High Pressure Loop with a design pressure of 26.2 MPa (3800 psi), and a design temperature of 360°C (680°F), and a flow of 1.26 to 5.05 l/sec (20 to 80 gpm) at high flux, could be modified or duplicated. Of course, testing of SCWR components, materials, cladding, and fuel would be accomplished with accurately controlled water chemistry, temperature, pressure, and flux that reflected the necessary test plan requirements.

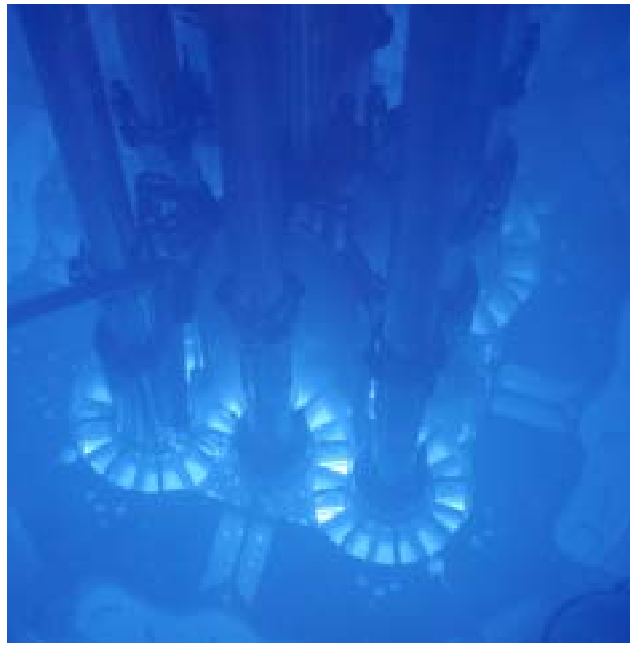

Figure 2. Photograph of the ATR core and loops.

Out-of-pile facilities for corrosion and/or SCC studies exist or are under construction at the Massachusetts Institute of Technology (MIT), University of Michigan, and University of Wisconsin. The Wisconsin facility, which can be used for both thermal-hydraulic studies and corrosion measurements, is described in Section 3.1.3 (Safety) below. Figure 3 presents a schematic representation of the current SCW facilities at MIT, which were originally developed for waste treatment studies. The exposure facility incorporates a relatively large autoclave with an internal volume of approximately 860 mls. It is large enough to expose a rack of samples (weight loss, welded, u-bend) for extended times. The high-pressure liquid chromatograph pump is capable of a maximum flow rate of 100 mls per minute. The SCW loop at MIT is being employed in a DOE Nuclear Energy Research Initiative (NERI) project to carry out corrosion experiments on candidate alloys in flowing SCW. The following materials have been identified for testing: ferritic/martensitic stainless steels (e.g., HT-9, T-91,

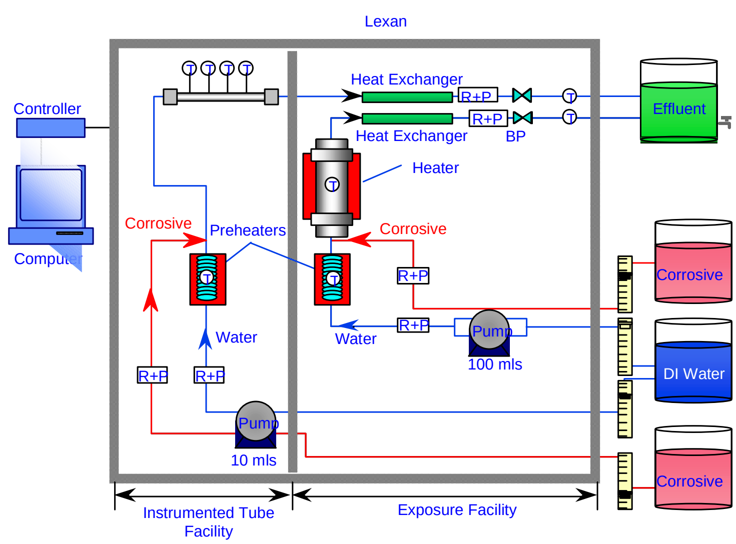

Figure 3. Schematic of the MIT SCW loop.

HCM12A), austenitic stainless steels (e.g., 304, 316), austenitic nickel-base alloys (e.g., Inconel 600 and 690), and precipitation-hardened nickel-base alloys (e.g., Alloys 718 and 625)

The University of Michigan facility is shown in Figure 4 and designed for both corrosion and SCC studies. In this loop system, one tensile sample can be tested in various loading modes such as CERT, constant load, ramp and hold, low cycle fatigue, etc. Additionally, 6 U-bend samples can be loaded into the test vessel, using sample holders secured to the vessel internal support plate. Water chemistry control includes the control of conductivity, pH, dissolved oxygen content, and the concentration of specific chemicals. The main pump controls the flow rate of the circulating water. Water from the test vessel passes a filter and ion exchanger where the corrosion products or any undesired contaminant are removed. During tests, the water conductivity and dissolved oxygen content are monitored at the inlet and outlet of the test vessel. The water is periodically sampled from the drain line of the main column to measure the pH. Mechanical loading is achieved using a stepper motor attached to the load frame. A tensile sample is connected to the motor through a pull rod on which a linear voltage displacement transducer and a load cell are installed. The facility will start operation shortly and be testing the same materials as are being tested in the MIT facility.

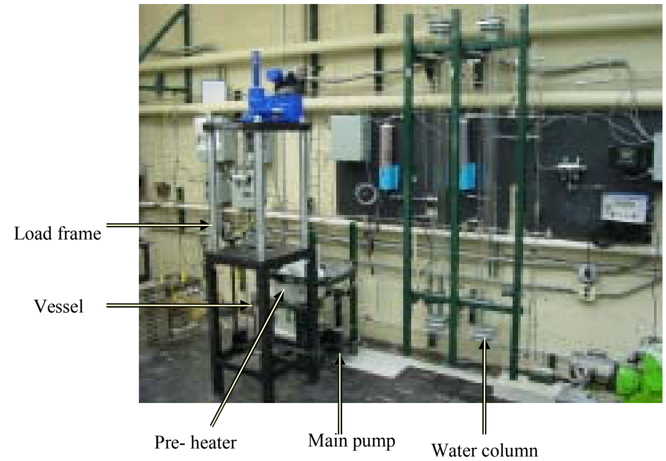

Figure 4. Overall view of the supercritical water loop system at the University of Michigan.

Neutron irradiations are essential to evaluate and qualify materials for Generation IV systems. However, much can be gained in the understanding of neutron irradiation effects using ion-beam facilities. Currently DOE-NE is sponsoring work using ion-beam facilities at both the University of Michigan (shown in Figure 5) and PNNL. These facilities provide excellent capabilities for studying microstructural and microchemical changes during irradiation as well as corrosion and mechanical properties in many environments. The higher dose rates must be taken into account and the depth of penetration is typically not sufficient to assess bulk mechanical properties. Yet, charged particle irradiations can provide a low-cost method for conducting valuable radiation effects research in the absence of, or as a precursor to verification experiments in reactors. While these facilities possess the

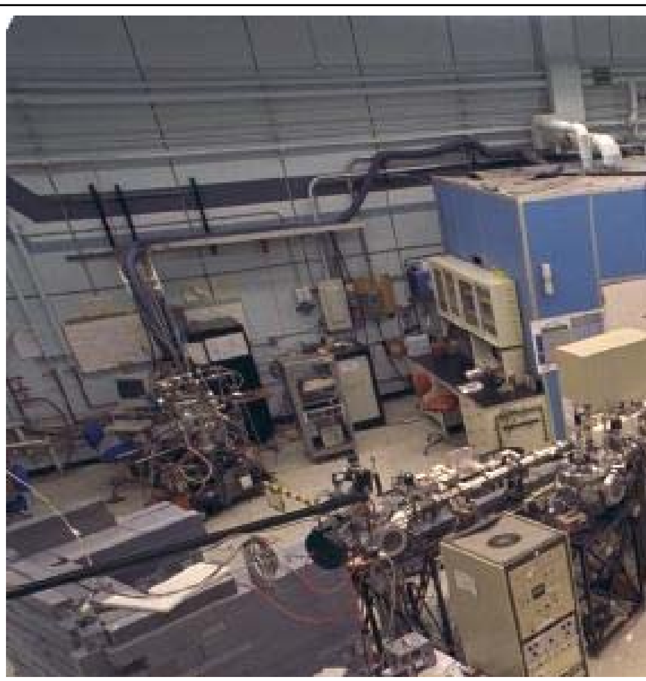

Figure 5. DOE-NE sponsored work using ion-beam facilities at both the University of Michigan and PNNL.

capability to study radiation effects on modest-sized programs, larger, more versatile facilities will be required for a major alloy design program such as that anticipated for the SCWR core internal materials.

Japan. Uniform corrosion and SCC tests are being performed in SCW loops at Toshiba and Hitachi. The Toshiba test section and associated loop, shown in Figure 6, were assembled to simulate SCW conditions up to 30 MPa, 600°C. Water chemistry can be controlled in terms of dissolved oxygen, dissolved hydrogen, and conductivity. The weight of the test specimens is measured to evaluate the uniform corrosion and the oxide films on the specimens are analyzed in terms of thickness, morphology, and chemical composition. SCC susceptibility is being examined by two means: double U-bend tests and slow strain rate tests. After testing, the double U-bend specimens are cut along their longitudinal centerline and the crack depths are measured. The stress and strain of the slow strain rate specimens are measured continuously during the tests. After the SCW exposure, the fracture surfaces of the specimens are inspected with a scanning electron microscope to identify the fracture mode. Also, the SCC area ratio on the fracture surface is measured to evaluate the relative SCC growth rate.

The test loop at Hitachi, Ltd. was also designed to evaluate the corrosion resistance of various alloys in simulated SCWR core water conditions up to 600ºC and 30 MPa. Dissolved oxygen can be controlled from about 10ppb to about 30ppm. H_2O_2, which will be generated as a radical during irradiation, can be added to the test section. Changing the DO and H_2O_2 concentrations, temperatures, and pressures can create various kinds of simulated SCWR core conditions. Both general corrosion and SCC tests are being carried out. Coupon type specimens are used for the general corrosion tests (30 specimens can be installed in the test section). U-bend or double U-bend specimens are employed for the constant strain SCC studies. The test loop can accommodate 15 SCC specimens at once.

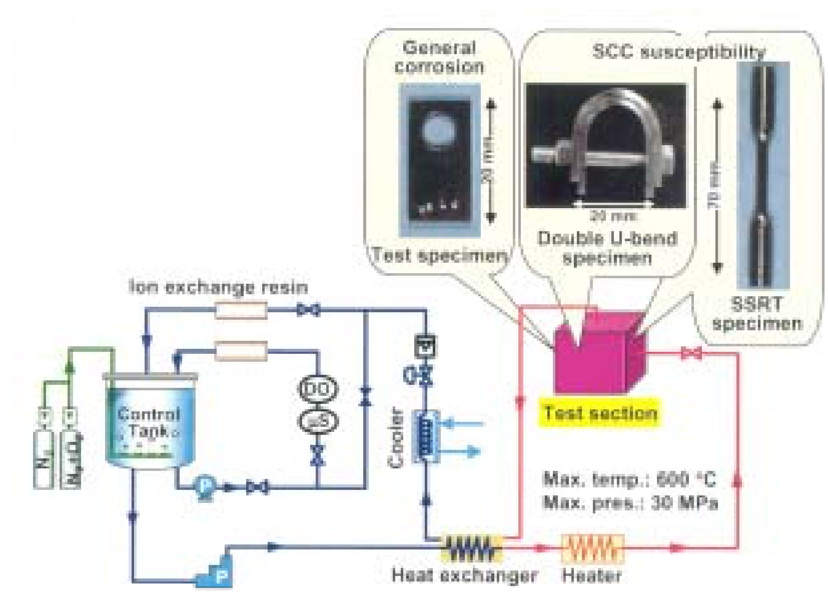

Figure 6. Supercritical water test loop at Toshiba.

The fundamental aspects of the SCWR water chemistry must also be investigated including the thermodynamics, electrochemistry, and radiolysis of the water at elevated temperatures. It is also important to understand the behavior of the corrosion products in high-temperature SCW because the radioactive corrosion products generated in a SCWR core may build up in the turbine area. Considering those aspects, the Japanese researchers have started gathering information on SCW chemistry from a variety of industries.

3.1.3 R&D Plan for SCWR Safety

3.1.3.1 Current State of Knowledge

Nuclear power safety initially can be achieved by reliable operation of nuclear power plants with a minimum number of abnormal events. This is a good starting point for defining safety because it points out that safety is rooted in reliable operation of the technology, which naturally prevents accidents. However, one cannot focus exclusively on reliable operation and accident prevention. The natural complement to reliability and prevention through outstanding normal plant operation is a comprehensive hazards analysis of the nuclear power plant, associated with mitigation of accident consequences.

Nuclear power has two inherent attributes, which make this a unique challenge. All technologies rely to some degree on the ability to empirically test that the integrated engineering system is 'safe' under a specified set of postulated accident conditions. Such testing is most usually done under full-scale conditions where a prototype might be destructively tested; e.g., automobile-crash testing for passenger safety. However full-scale experimentation of nuclear power plant accidents is technically and economically impractical. Even in the commercial airline industry there has only been one destructive crash test of a full-scale airliner in decades. The second attribute is that reactor accidents have the potential to release large amounts of radioactivity into the environment, but uncontrolled releases of radioactivity into the environment have been (and must be) very rare events. Three Mile Island and Chernobyl have occurred in the last 23 years. However, by now over 10,000 reactor-years of commercial experience has been accumulated worldwide. Continued vigilance is required to help to assure that severe accidents will never occur again.

The concept of defense in depth is fundamental to the safety of nuclear installations (INSAG-10 1996). The strategy is twofold: first to prevent accidents and, second, if prevention fails, to limit their potential consequences and prevent any evolution to more serious conditions. Defense in depth is generally structured in five levels. Should one level fail, the subsequent level comes into play. Level 1 is prevention of abnormal operation and system failures. This is achieved by conservative design and high quality in construction and operation. Level 2 is control of abnormal operation and detection of failures. This is achieved by control and protection systems and by other surveillance features. Level 3 ensures control of accidents within the design basis, and is achieved by activating engineered safety features and accident procedures. The measures taken at this level are aimed at preventing core damage. Design and operating procedures are aimed at maintaining the effectiveness of the barriers, especially the containment, in the event of such a postulated accident. The third line of defense consists of incorporating specific plant engineering safety features that effectively mitigate a range of postulated abnormal occurrences or accidents. These safety systems may require active equipment actuation or may use passive or natural processes such as gravity-induced flows. These postulated accidents then become part of the nuclear plant design base for safe operation under normal and anticipated accidents; i.e., DBAs. Level 4 involves control of severe plant conditions including prevention of accident progression and mitigation of the consequences of a severe accident. The containment structure and associated systems are particularly important severe accident mitigation devices. Level 5 involves mitigation of radiological consequences of significant releases of radioactive materials by off-site emergency measures.

A natural result of identifying these multiple lines of defense is to understand the types of accidents that these design features (and physical barriers) protect against. In addition, each type of accident has a certain probability of occurrence, which becomes more rare as its severity increases. As the original NRC Reactor Safety Study (WASH-1400, 1975) stated, the performance of these multiple lines of defense should be continually assessed by use of probabilistic risk analysis as well as deterministic engineering calculations. For our current discussion, we focus on the third and fourth lines of defense; i.e., the inherent differences for DBAs and severe accidents for the SCWR designs. Note that we do not address the subject of thermal-hydraulic-neutronic instabilities in this section of the research plan; that material is in Section 3.1.4 below.

Japanese researchers (Ishiwatari et al. 2002, Kitoh et al. 2001, Lee et al. 1998) have been the first group to perform safety analyses for the SCWR designs (Oka et al. 2002, Oka and Koshizuka 2000), both thermal and fast. Because of the significant differences between some of the phenomena in SCWRs and in the current LWRs, the Japanese needed to reconsider some of the criteria normally used to assess reactor safety. The transient and accident safety criteria related to fuel design, the reactor core, the primary system, and the containment were assessed, and in some cases revised. Specifically, their analyses suggested that the key damage criterion should be a fuel cladding temperature limit (the actual value depends on the fuel cladding material composition) for transient and accident conditions. Note that the phenomenon of critical heat flux (CHF) is not present above the critical pressure and thus the fuel cladding temperature becomes the limiting criterion that must be considered during postulated transients and accidents. This design limit also exists under steady-state conditions, as does the condition to limit fuel pin linear power to avoid fuel melting.

Kitoh et al. (2001) analyzed 12 DBA and more frequent plant transients for two versions of their fast spectrum SCWR designs and concluded that the SCWR designs can indeed satisfy a fuel cladding temperature criterion of 1260ºC during DBAs, and satisfy a criterion of 610ºC (for Type 316 stainless steel cladding) or 840ºC (for nickel-based cladding) during expected transients. Ishiwatari et al. (2002) analyzed a similar variety of accidents and transients for their thermal spectrum SCWR design and concluded that the hottest cladding temperature would occur during a total loss of feedwater accident and the maximum cladding temperature would be about 1010ºC (well below the 1260ºC criterion for stainless steel previously recommended by the U.S. NRC). The peak cladding temperatures during expected transients calculated by Ishiwatari et al were also reasonably below their criterion of 840ºC for nickel-based cladding. The Japanese researchers also evaluated the consequences of reactivity-initiated accidents (RIAs) and found that the energy inserted in the fuel rods would be within currently accepted values.

It should be emphasized that these analyses are based on extrapolations of past data as well as extrapolations of LWR models. Thus, it is our belief that additional experimental data is required, along with improved modeling capability to verify these judgments. These new data and models would form the technical bases for assuring acceptable safety for SCWR systems for postulated DBAs as well as mitigation for severe accident phenomena.

These design and research activities provide the major background information for safety investigations regarding SCWRs. Based on a review of this design (Oka and Koshizuka 2000, Oka et al. 2002) and associated analyses, the SCWR is assumed to have the following design features:

- Fuel rod design parameters similar to an LWR in composition and geometry (although cladding materials may be somewhat less susceptible to exothermic oxidation)
- Neutronic core design attributes that have higher fuel enrichments and various amounts of moderation (this would be affected by the type of moderation and associated neutron spectrum)
- Coolant conditions above the water critical pressure and spanning the water critical temperature (possibly from 550K to 900K) with water chemistry that is expected to be similar to LWRs
- ECCS and containment systems that are quite flexible in design concept; i.e., similar to current active safety systems like those for the ABWR or similar to the passive safety systems envisioned for the ESBWR or the AP-1000
- DBAs and potential accident initiators for severe accidents similar to LWRs; i.e., no expected qualitative differences that are significant from LWR hazard analyses.

3.1.3.2 Gaps between What We Know and What We Need to Know

The design features discussed above suggest that the SCWR system temporal response will probably be the key determinant in the evolution of the accident, whether the accident is a DBA or a severe accident. The time-scale of the accident would probably differ from current LWR systems, and this must be taken into account in the SCWR safety analyses. Some of the phenomena that could affect the time-response of the SCWR system and are not well known include:

1. The coupled effects of the neutronics/thermal-hydraulics during DBAs, as well as any implications from fuel rod composition or geometry for severe accidents.
2. Transient flows (e.g., choked-flow) and heat transfer phenomena, particularly near the critical region and how the transient would evolve into the two-phase region or the single-phase gas region.
3. Power-flow instabilities that may be induced during the transient.

The gaps in our understanding of the response of SCWR systems and components during normal, off-normal, and accident conditions can be divided into a few topics:

- Phenomena that require more fundamental study due to the novel nature of using water above its critical point as a coolant;
- Evaluation and analysis of the SCWR system design to determine the transient behavior and system interactions for all the DBAs of interest;
- Extensive design studies and associated optimization of the SCWR system to make the plants as safe as practical.

Consider each topic area separately.

SCWR Phenomena

Successful development of SCWRs depends on a transparent demonstration of their safety performance. However, existing LWR thermal-hydraulic models, correlations and codes are not validated for supercritical conditions. For example, the heat transfer at supercritical pressures differs strongly from that at subcritical pressures due to the large variation in the properties near the pseudo-critical temperatures. Also, at supercritical pressures a deterioration of the fuel to coolant heat transfer occurs at high values of the heat flux and/or low values of the coolant flow rate. The exact mechanism of this so-called *relaminarization* phenomenon is not well understood. LWR single -phase heat transfer correlations (e.g., Dittus-Boelter) do not correctly reproduce the supercritical data. Single tube supercritical fluid heat transfer experiments have been conducted and correlations developed using that data, however, all these correlations were derived for circular tubes. Application of these correlations to rod bundles is uncertain. Also, the basic thermal-hydraulic properties of the SCW, particularly SCW transport properties such as thermal conductivity and dynamic viscosity, are somewhat more uncertain (up to 10% uncertainty) near the pseudo-critical temperatures than the properties normally used to evaluate LWR transients (about 2% uncertainty).

Also, the concept of choked flow near or above the critical point may be quite different. For any given supercritical pressure, there exists a temperature (called the pseudo-critical temperature) at which the speed of sound exhibits a minimum, which has the potential to affect the behavior of a supercritical reactor during a LOCA, other depressurization transients, or for the design of key ECCS components. For example, because the speed of sound is so low at the pseudo-critical temperature, choked flow might indeed be established at this condition. In this case, the flow at the break would be single-phase, which is unusual in LWR systems. Also, the sound-speed minimum corresponds to a relatively low fluid density. Therefore, the choked flow, which is the product of velocity and density at the break, could also be very low. This would significantly change the time constant of the system early in the accident event, e.g., for given size of the break, SCWR might depressurize at a considerably lower rate. However, there is no experimental information on the critical flow of SCW to verify this possible behavior.

Simulation of SCWRs is made inherently more complicated by the large variation of the thermodynamic and transport properties over the pressure and temperature range of interest. LWR safety and performance analysis codes, which make use of traditional heat-transfer and choked-flow models and correlations not proven for SCWRs, will need to be reevaluated.

In addition to the thermal-hydraulic phenomena where there are significant gaps in our information, data, and understanding, there are some issues associated with the behavior of the fuel during DBAs that must be addressed. Probably the most important issue is the ballooning behavior of stainless steel or nickel-based alloy clad fuel rods during a LOCA. Although the ballooning behavior of Zircaloy clad fuel rods has been extensively measured and modeled, very little work has been done on fuel rods with other advanced cladding materials.

SCWR System Behavior

The overall system response of the SCWR should be analyzed for a range of transient sequences. Because the SCWR is a novel water reactor system with a new combination of neutronic and thermal-hydraulic parameters and system interactions, it is important to ascertain the characteristic times for various transients that the system would experience during normal, off-normal, and DBA operation, as well as serve as a design basis for hazard analyses. These transients could be categorized in three generic groups:

- Normal plant start-up, shutdown and other operational transients—e.g. load following transients;
- DBAs that have formed the basis for ECCS designs;
- Hypothetical severe accident sequences.

For each of these transients and accident sequences, the SCWR transient response needs to be determined as well as the neutronic/thermal-hydraulic stability response of the SCWR system.

SCWR Safety System Design and Optimization

The following are examples of SCWR safety system design topics to achieve competitiveness in economics without sacrificing safety/reliability.

- Reactor protection logic [SCRAM, ECCS, residual heat removal (RHR), containment, passive systems] and criteria: Because the SCWR operational states are different from LWRs, the safety and operational criteria should be reconfirmed under the full range of conditions.
- Reactivity control: Because the reactivity cannot be controlled with the re-circulation–flow as in commercial BWRs, or by boron-concentration-control as in PWRs, the reactor control system should be designed for high reliability. Also, the need for and design of any redundant reactor shutdown system needs additional R&D.
- Coolability: To minimize design margins for core cooling under normal as well as abnormal conditions, it is desirable to perform necessary heat transfer tests and thus to develop accurate and simple heat transfer models for SCW which incorporate geometrical heat transfer enhancing effects and flow characteristics including cross flow, stability and natural convection.

- Severe accident mitigation systems: To satisfy defense-in-depth, severe accident mitigation systems should be considered in the SCWR. Because most mitigation systems developed for LWRs are applicable to SCWRs, design optimization of such systems is essential. Accurate hydrogen generation rates due to high-temperature water-metal reaction would facilitate the establishment of system design margins.
- In-core sensors: Because thermal-hydraulic conditions for SCWRs are different from those for LWRs, attention needs to be paid to in-core sensor designs.
- Plant parameter control systems: Because thermal-hydraulic conditions as well as plant responses for SCWRs are different from predecessors', plant parameter (pressure, temperature, flow rate, etc.) control systems need to be redesigned.

3.1.3.3 Proposed R&D Program

We envision a SCWR safety research program organized around the following topics:

- Reduced uncertainty in SCW transport properties.
- Further development of appropriate fuel cladding to coolant heat transfer correlations for SCWRs under a range of fuel rod geometries.
- SCW critical flow measurements, as well as models and correlations.
- Measurement of integral LOCA thermal-hydraulic phenomena in SCWRs and related computer code validation.
- Fuel rod cladding ballooning during LOCAs.
- SCWR safety design optimization studies including investigations to establish the effectiveness of passive safety systems reactivity shutdown systems.

The purpose of making additional *basic thermal-hydraulic property* measurements at and near the pseudo-critical temperatures would be to improve the accuracy of the international steam-water property tables. This work could be done at a university or at the National Institute of Science and Technology over a 3- to 5-year time frame for a total cost of about $5 million.

The fuel cladding to SCW *heat transfer* research should consist of a variety of out-of-pile experiments starting with tubes and progressing to small and then relative large bundles of fuel rods. The bundle tests should include some variations in geometry (fuel rod diameter and pitch, bundle length, channel boxes, etc.), axial power profiles, coolant velocity, pressure, grid spacer design, etc. The larger bundle tests will require megawatts of power and the ability to design electrically heated test rods with appropriate power shapes. It is expected that this program might take 5 to 6 years and cost of about $2.5 million per year.

The SCW *critical flow* experiments would be separate effects out-of-pile experiments with variations in hole geometry and water inventory. This research would take about 4 to 5 years and cost on the order of $1.5 million per year.

The *integral SCWR LOCA thermal-hydraulic experiments* would be similar to the Semiscale experiments previously conducted at the INEEL for the U.S. NRC to investigate LOCA phenomena for the current LWRs. We would probably need a full height system and a large enough core diameter for appropriate scaling. A Semiscale type test program and the related computer code development would take about 10 years and cost on the order of $10 to 12 million per year. It may be possible to design this facility to accommodate the heat transfer research discussed above as well as the needed LOCA testing, and even some thermal-hydraulic instability testing (discussed in Section 3.1.4 below).

Fuel rod *cladding ballooning* is an important phenomenon that may occur during a LOCA. Although considerable work has been done to measure and model the ballooning of Zircaloy clad fuel rods during LOCAs, little is known about the ballooning behavior of stainless steel or nickel based alloy clad fuel rods during a LOCA. It is expect that this information could be obtained from out-of-pile experiments using fuel rod simulators. The research would take form 4 to 6 years and cost about $1 million per year.

SCWR *safety design optimization* is also an important area of needed research. It is likely that any Generation IV plant will need to be at least as safe as the current ALWR designs. In fact considerable work should go into finding ways to make SCWRs even safer than ALWRs. All of the known accident scenarios must be carefully evaluated (large and small bread LOCAs, RIAs, loss of flow, main steam isolation valve closure, over cooling events, anticipated transients without scram, high and low pressure boil off, etc.) to assure compliance with reactor protective criteria. There may be safety features, such as redundant reactivity shutdown systems, that require special designs.

Assuming that the SWCR has passive systems similar to the ESBWR and/or the SWR-1000, the PANDA facility at the Paul Scherrer Institute (Switzerland) could be used to confirm the performance of these systems and to investigate the factors that influence the reliability of these passive systems. PANDA is a large-scale thermal-hydraulic test facility that allows investigation of PCCSs and long-term containment behavior after a LOCA. Other facilities that may be useful in examining passive systems for injecting water into the primary system include Japan Atomic Energy Research Institute's (JAERI's) ROSA-LSTF facility and the SPES integral test facility of SIET, Piacenza, Italy. Test facilities for investigating basic phenomena (e.g., natural circulation) on which passive systems are based exist at several institutes, for example, in Italy, the UK, Russia, India, Finland, Germany, the Republic of Korea, the Netherlands, and elsewhere. It is estimated that tests to confirm the performance of passive systems, to investigate the factors which influence their reliability, and to investigate certain basic phenomena on which passive systems are based can be conducted for a total of approximately U.S. $7–10 million, over a period of 3–5 years. The analyses to optimize these systems might require another $1 to 2 million per year.

Existing Facilities and Expertise

A brief overview of the facilities in participating countries is provided in this section. In the past virtually all work was performed at prototypical LWR facilities and at prototypic conditions. Most existing facilities were not designed and licensed for operation at the higher temperatures and pressures required for SCWRs. Therefore, in general it will not be possible to utilize these facilities for R&D of the SCWRs, particularly at nominal reactor operating conditions. However, in order to make efficient use of the expertise and resources already available, the organizations that performed LWR safety experiments and analyses in the past, could also be involved in SCWR safety research in the future, as well.

United States. The SCW flow loop shown in Figure 7 is being deployed at the University of Wisconsin at Madison for a current NERI project. The loop is electrically heated, relies on natural circulation of the supercritical fluid and can be operated at pressures up to 30 MPa and temperatures up to 600°C, which allows conducting experiments above the critical pressure of water and on either side of the critical temperature. Test-section velocities are designed to be up to 1 meter/sec and can be achieved with an input energy of up to 80kW. Although the loop will be initially used for corrosion studies, it can also be used for thermal-hydraulic studies (with small rod bundles), as well, in particular for natural circulation instabilities. Also as part of this NERI work, Argonne National Laboratory is developing a supercritical carbon dioxide natural circulation loop to specifically examine instability behavior. This loop can be operated up to 10 MPa and at temperatures up to 150ºC with flow velocities designed to be about 1 meter/sec. Finally, a critical flow blow-down facility is being designed at the University of

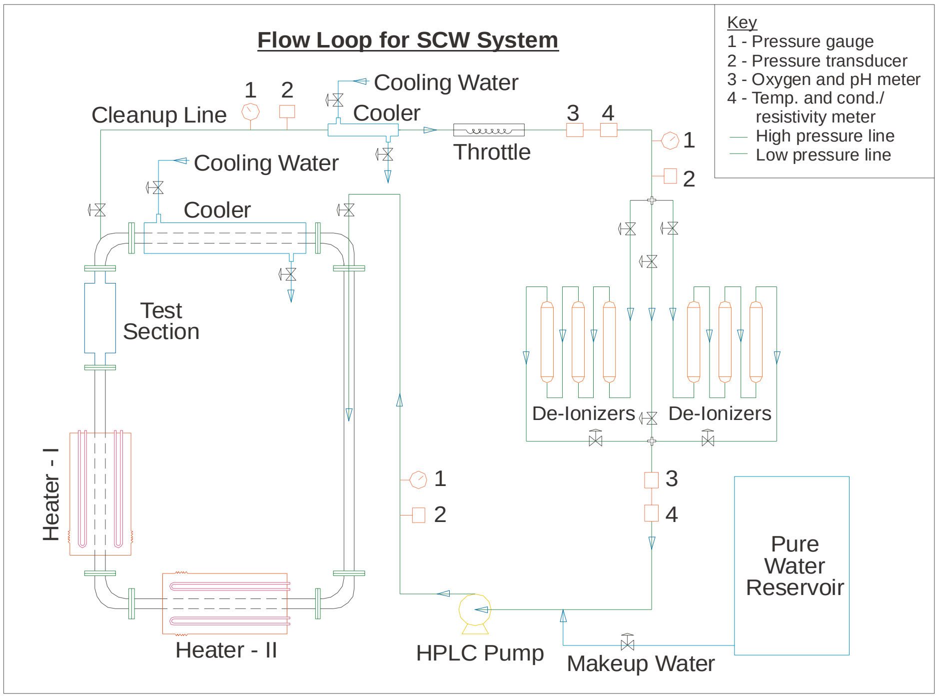

Figure 7. University of Wisconsin supercritical water test loop.

Wisconsin-Madison for SCW with pressures up to 30 MPa and temperatures up to 600°C. The facility will be able to determine multiphase flow properties during the transient.

Japan. Kyushu University has a supercritical Freon (R22) loop, which has been used to measure the heat transfer at SCW fossil boiler tube surfaces. (The pseudo-critical pressure of R22 is 5 MPa, and the pseudo-critical temperature is 96°C.) The maximum mass flow rate of the facility is about 2000 kg/m^2/s; and the maximum heat flux is about 200 kW/m^2, which allows for measurements of single-phase heat transfer and heat transfer deterioration. Currently a Japanese team is conducting R22 tests at Kyushu University to validate heat transfer equations for various geometries, to develop heat transfer enhancing factors, and to study heat transfer under flow transient conditions. Pressure drop correlations are also being validated for supercritical fluids. The University of Tokyo owns a supercritical CO_2 loop that can be used to visually observe supercritical fluid behavior. JAERI and Toshiba also own full-height test facilities for testing various PCCS designs. NUPEC (a governmental organization) owns a large shaking table to examine system reliability under seismic conditions.

Canada. AECL operates a loop for supercritical carbon dioxide thermal studies. The loop can run at a pressure in the 2.8–10 MPa range and at temperatures up to 310°C, with mass flow rates for from 0.05 to 2 kg/s. The power is delivered to the test section by a 350 kW power supply. The loop has a steel enclosure, which permits the use of unregistered pressurized test sections. This facility can be used to support SCWR R&D activities in the following areas:

- Development of heat transfer correlations, friction coefficients and flow instability under SCW conditions
- Establish fluid-fluid modeling criteria for interpreting results from the supercritical pressure CO_2 facility
- Development of a coupled thermal-hydraulic-neutronic code.

There are also two facilities that will be built in the next few years. The first is a small/intermediate natural circulation loop for SCW, specifically designed to investigate instabilities of the buoyancy type. This work has already started through a collaborative program with the University of Manitoba with joint AECL and Canadian Foundation for Innovation funding. This facility can be used to support SCWR R&D activities in the following areas:

- Development of heat transfer correlations, friction coefficients and flow instability under SCW conditions,
- Development of coupled thermal-hydraulics-neutronic codes,
- Development of power envelopes for pumped and natural-circulation flow,
- Development of water chemistry and material specifications for heat transport and feed-water systems, and
- Development of water chemistry specifications for both indirect and direct cycles.

The second facility is a small loop to study passive safety systems (with emphasis on passive moderator cooling). This facility can be used to support SCWR R&D activities in the following areas: (a) development of coupled thermal-hydraulics-neutronic codes, and (b) development of power envelops for pumped and natural-circulation flows.

Europe. To the best of our knowledge, no facilities for SCWR safety studies are currently available in Europe. The Framatome ANP Benson Test Facility in Erlangen, Germany has been used for various heat transfer studies including testing of supercritical boilers tubes for fossil-fired plants. Its characteristics (a maximum pressure of 33MPa, a maximum temperature of 600°C, a flow rate of 28 kg/s, and a power of 2000 kW) allow for heat transfer experiments in bundle geometry.

Russia. The Institute of Physics and Power Engineering (IPPE), in the Russian Federation, has a number of large SCW test facilities as well as a Freon facility. The Russian capacities are shown in Table 6 below.

Table 6. Supercritical fluid thermo-hydraulic test facilities at IPPE.

Name	Description	Fluid	Power (kW)	Press. (MPa)	Temp. (C)	Flow (t/h)
SVD-2	High-pressure 3 loops Bundle tests Steam generator tube test	Water	9200	26 20 10	500 365 310	35 35 10
SKD-1	Supercritical test loop Bundle tests	Water	1000	26	530	8
STF	2 loops Channel flow in various geometries	R12	1000	4	350	40

3.1.4 R&D Plan for SCWR Power-Flow Stability

3.1.4.1 Current State of Knowledge

SCWRs present the possibility of various types of instabilities, namely, density-wave instabilities, coupled thermal-hydraulic/neutronic instabilities, and natural circulation instabilities (including side-to-side instabilities). Each of these types of instability is discussed in a little more detail below.

Because of the large axial variation of the water density at supercritical pressure, SCWRs are potentially vulnerable to density-wave instabilities. Due to the compressibility of the coolant, flow rate perturbations travel at a relatively low speed in the core. Therefore, any perturbation (e.g., an increase of the flow rate at the core inlet) is not instantaneously dampened by the friction and form losses throughout the core. Instead, relative phase lags of the different components of the momentum equation (e.g., inertia, acceleration, gravitation, friction, form, pressure) are generated, which can lead to self-sustaining oscillations resulting in fuel failure from overheating and/or thermal cycling.

Furthermore, because of the effect of the coolant density on the neutron flux (power) in the SCWR core, there is the potential for coupled thermal-hydraulic/neutronic instabilities. An increase in density increases the flux/power and this in turn can decrease the density, which in turn decreases the flux/power leading to an increase in density. If these changes are dampened or if there are safety measures introduced (e.g., movement of control rods), then there should be no problem. However, if these oscillations grow, then there is the potential for fuel damage, and this potential can be exacerbated if safety measures are not taken or are delayed.

Finally, if the reactor operates as a natural circulation loop during certain loss-of-flow transients (and under normal operating conditions for some designs), there is the possibility for buoyancy oscillations, in which there exists a phase lag between the density change, the coolant flow, and the pressure drop terms of the loop momentum equation.

The oscillations can be either in-phase (core-wide) where all the bundles oscillate in phase or they can be out-of-phase or regional oscillations where the oscillations in different spatial regions (e.g., one half of the planar core) are out of phase. However, with respect to the direct cycle designs, the likelihood of this latter mode of oscillation might be reduced because the SCWR fuel assemblies generally do not need to be separated into parallel channels.

It is necessary for any given design to show that either the oscillations do not occur during normal operation or that if they do, they can be detected and suppressed in a safe manner. Note that normal operation includes modes other than full power, e.g., startup and normal runbacks of power. Different start-up approaches are possible including constant-pressure and a sliding-pressure start-up similar to the common practice in coal-fired supercritical power plants. Each of these start-up approaches will likely present unique instability issues in a nuclear reactor as the power-flow envelope encompassed differs from approach to approach. Finally, oscillations under accident conditions must also be considered, e.g., under anticipated transient without scram conditions.

Another important issue is that of control of the main reactor variables, e.g., core power, coolant pressure and temperature. The University of Tokyo group designed a control system for their direct-cycle fast reactor where the core power is controlled by the control rods, the pressure by the turbine throttle valve, and the coolant core outlet temperature by the pump flow. However, other approaches are possible and/or necessary depending on the specific design, e.g. in an indirect-cycle SCWR a pressurizer might be needed to control the pressure, in a thermal reactor the power would likely be controlled by the feed-water flow rate, etc.

3.1.4.2 Gaps between What We Know and What We Need to Know

There has been a considerable amount of research done in the area of instabilities for the lower pressure and temperature conditions typical of BWRs, as well as some analytical work in Japan for the SCWR. The BWR experience is relevant in that it helps point the way toward what needs to be repeated for the SCWR-T. It is interesting to note that although BWRs were originally designed ~50 years ago, the past 10 years has seen considerable research activity in stability analysis in order to be able to answer problems that continue to concern the BWR plant operators.

However, experimental work aimed directly at simulating the thermal-hydraulic, coupled neutronic/thermal-hydraulic and buoyancy instability performance at the operating and off-normal conditions of SCWRs has not been performed. Furthermore, the heat-transfer, pressure-drop and critical-flow correlations, models and codes used for safety analysis (including instability analysis) of LWRs are not validated for supercritical conditions.

In the following R&D activities only the neutronic and thermal-hydraulic behavior is considered; no consideration is given to fuel behavior as a result of oscillations.

3.1.4.3 Proposed R&D Program

The objective of this R&D is a better understanding of instability phenomena in SCWRs, the identification of the important variables affecting these phenomena, and ultimately the generation of maps (a conceptual example is shown in Figure 8) identifying the stable operating conditions of the different SCWRs designs. Consistent with the U.S. NRC approach to BWRs licensing, the licensing of SCWRs will probably require, at a minimum, demonstration of the ability to predict the onset of instabilities. This can be done by means of a frequency-domain linear analysis. Prediction of the actual magnitude of the unstable oscillations beyond onset, although scientifically interesting and relevant to beyond-design-basis accidents, will likely not be required for licensing and can be delayed to a second phase of the SCWR development. This will simplify the R&D program and will result in significant cost savings.

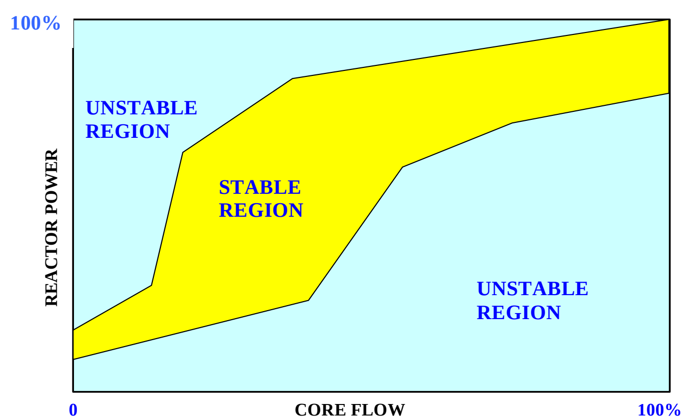

Figure 8. Conceptual envelope for stable power-flow operation of SCWR.

Both analytical and experimental studies need to be carried out for the conditions expected during the different operational modes and accidents. The analytical studies can obviously be more extensive and cover both works in the frequency domain as well as direct simulations. These studies can consider the effect of important variables such as axial and radial power profile, moderator density and fuel temperature reactivity feedback, fuel rod thermal characteristics, coolant channel hydraulic characteristics, heat transfer phenomena, core boundary conditions (including the effect of direct or indirect cycles), etc. Mitigating effects like orificing, insertion of control rods, and fuel modifications to obtain appropriate thermal and/or neutronic response time constants can also be assessed using analytical simulations. Computer models that are used in this analysis must be validated using results from experiments. Specifically, the heat-transfer, pressure-drop, and critical-flow correlations and models used in LWR codes must be upgraded or replaced with models developed for prototypical SCWR conditions.

Extensive analytical work is also needed in devising adequate strategies for control of the main reactor variables for the different specific SCWR designs.

Instability experiments to be considered include both with and without nuclear heating with the former being much more difficult. Experimental loops without nuclear heating would be expected to supply most of the data on instabilities with superheated water and at other conditions that might be possible for a given SCWR design. Since experiments with nuclear heating are difficult and expensive, most or all of the stability research with nuclear feedback might be done with validated computer codes.

If instability is determined to be a problem then instrumentation and the means for precluding getting into a region of instability and for getting out of the region must be examined.

R&D Linkages/Dependencies

Strong synergies exist with the investigation of basic heat-transfer and critical-flow phenomena in SCW described elsewhere in this report. Correlations and models developed for steady state or transient analysis of the SCWRs will be used for stability codes, as well. Analogously, facilities designed to investigate thermal-hydraulic instabilities can also produce basic data on thermal and flow phenomena.

Existing SCW Facilities and Expertise

In this section a brief overview of the expertise and facilities in GIF countries in the area of nuclear reactor instabilities is provided. In the past virtually all work was performed at prototypical LWR conditions. Most existing facilities were not designed and licensed for operation at the high temperatures and pressures of the SCWRs. Therefore, in general it will not be possible to utilize these facilities for R&D of the SCWRs. However, in order to make efficient use of the expertise and human resources already available, the organizations that performed LWR instability analysis and/or experiments in the past, should probably be involved in the instability R&D for the SCWRs in the future.

Japan. There is no SCW flow loop to investigate instability phenomena in Japan. However, in the past three decades considerable experimental and analytical research was performed on stability of BWRs (7 MPa. pressure facilities). Core instabilities, regional instabilities, and channel thermal-hydraulic instabilities are all important in current BWR designs.

Multichannel thermal-hydraulic instability tests have been performed at the TOSHIBA-BEST and NFI-Test Loop facilities since about 1980. The BWRs cores are composed of many parallel channels and the pressure difference between lower and upper plenum is kept constant. In order to simulate channel thermal-hydraulic instabilities, both test facilities consisted of a 3 x 3 test bundle simulating the high power channel and 8 x 8 or 6 x 6 parallel bundles maintaining the pressure difference constant across the core. Many tests were performed to evaluate the stability characteristics for new fuel geometries such as the 9 x 9 bundle, the effect of geometry on the stable limiting power, the thermal-hydraulic behavior beyond the stable critical power conditions, and the channel thermal-hydraulic instability behavior with the neutron kinetic feedback.

Recently, coupled neutronic/thermal-hydraulic instability tests were performed using the THYNC loop at the JAERI. This test facility consisted of parallel channels and featured a rapid simulation of void reactivity using instantaneous channel cross-section void measurements and a computer-controlled simulated rod power change method. The thermal-hydraulic instabilities of natural-circulation BWRs were also studied with the SIRIUS loop at CRIEPI.

Canada. AECL is in the process of deploying a natural circulation loop for SCW, specifically designed to investigate instabilities of the buoyancy type.

Europe. To the best of our knowledge no facilities for SCWR instability studies are currently available in Europe. At present there exists a European project, named NACUSP, which is investigating BWR instabilities in both forced-circulation (e.g., ABWR) and natural-circulation (e.g., ESBWR) designs. Four facilities are being used in this project:

- The CLOTAIRE facility at the CEA Cadarache research center is a large-scale Freon natural-circulation facility, which was used in the past to simulate the behavior of the secondary side of the French PWR steam generators, and has now been modified to simulate the re-circulation loop of a BWR;
- The DESIRE facility at Delft University also uses Freon and simulates the high-pressure operation of the Dodewaard natural-circulation BWR;
- The CIRCUS facility also at Delft University is a low-pressure (0.1–0.5 MPa) water/steam 1:1 height-scaled loop of the Dodewaard reactor;
- The PANDA facility at PSI is a low-pressure (1 MPa) large-scale water-steam facility with an installed power of 1.5 MW.

Within the framework of this project, the EHTZ is producing a version of a linear stability analysis code capable of considering all channels in a BWR and 3D core kinetics.

ASEA in Sweden (later ABB, and now Westinghouse) has a one bundle loop for BWR stability analysis. In Italy CISE has done stability work for PWR steam generators with a Freon loop (the VAPORE facility), while the University of Pisa has the PIPER-ONE facility that was last used for stability work. Stability data have also been obtained in the past in the Dutch natural circulation BWR at Dodewaard, now shut down.

Schedule and Cost

It is envisioned that instability experiments will be conducted at the multi-purpose SCW thermal-hydraulic facility described in the Section 3.1.3 above. The test section will be designed to accommodate a single bundle as well as multiple bundles. This will enable studying in-phase and out-of-phase density-wave oscillations. Moreover, the facility will provide a natural circulation flow path for the coolant to study buoyancy loop instabilities. Assuming the construction cost of the facility is budgeted in the safety studies described in Section 3.1.3, it is projected that the instability experiments and related analytical work will cost about $3 million per year for a total of 3 to 4 years. Further work would depend on the issues uncovered during the experimental program.

The analytical work would include making sure that the codes are validated, doing the appropriate analyses, and repeating the experiments for different reactor designs. This program would be closely linked to other research activities validating the models used in the computer codes and applying to the codes to other issues such as transient/accident analysis. In conclusion, the instability issues could probably be resolved within a 4-year period for a cost of ~$12 million. This assumes that other complementary activities (measurements of constitutive laws, validation of codes, etc.) are taking place at the same time.

3.1.5 R&D for SCWR Plant Design

3.1.5.1 Background

SCWRs are being developed based on SCW fossil power plant technologies as well as reliable LWR technologies. The SCW fossil power plants have been successfully operated for more than 30 years in the world. In addition, more than 300 LWRs are operated in the world; their world-average capacity factor has exceeded 80% in 1998-2000.[b] The development principle for the SCWRs should be maximum utilization/adoption of such technologies with minor modifications.

3.1.5.2 Gaps

Because many of the major systems that can potentially be used in a SCWR were developed for the current BWRs, PWRs, and SCW fossil plants with conservative criteria and assumptions, the major plant design and development needs that are unique for SCWRs are primarily found in their design optimization as well as their performance and reliability assurance under SCWR neutronic and thermo-hydraulic conditions different from the current LWRs. Two major differences in conditions are the stresses due to the high SCWR operating pressure (25 MPa) and the large coolant temperature and density change (approximately 280 to 500°C or more, 800 to 80 kg/m^3, respectively) along the core under the radiation field.

3.1.5.3 R&D activities

Design Optimization Study

The following are examples of design features that need to be optimized to achieve competitiveness in economics without sacrificing safety or reliability. This work is expected to take about 8 to 10 years and cost about $120 million.

Reactor systems:

- Fuel assembly design: Because of large density change along the core, the moderator must be designed to minimize the fissile enrichment needed for the thermal spectrum SCWRs.
- Control rod design: Because reactivity can not be controlled by re-circulation flow control like in BWRs, or by boron-concentration-control like in PWRs, the control rod (and CRDM) must be designed for highly reliable operation. In addition, the coolant temperature drop due to control rod insertion should be minimized by design and operational procedure so as not to decrease the power and the thermal efficiency.
- Internals (including fuels, control rods): Because of the large difference between the coolant inlet temperature and outlet temperature, the internals must be designed to minimize the thermal stress, thermal fatigue, and heat transfer from the hot to cold coolant (in the case of the thermal spectrum SCWR with water rod moderation) under normal as well as transient conditions.
- Reactor vessel: The nozzle should be designed for a pressure of 25 MPa. The internal surfaces of the reactor vessel should be designed to minimize corrosion.

Main steam systems:

- Pressure relief systems: It is necessary to develop critical flow models for SCW as well as direct contact heat transfer models between SCW and subcritical pressure water (the critical flow research

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b. IAEA PRIS: http://www.iaea.org/programmes/a2/

is discussed in Section 3.1.3). The critical flow models and the direct contact heat transfer models will facilitate the design of the pressure relief systems.

- Coolant clean-up systems / Coolant chemical control systems: The current LWR clean-up, as well as chemical control, systems are not designed for SCWR conditions and it will be necessary to develop new ones. To do this we will need to understand the thermodynamic properties of the radioactive elements in SCW systems. Iron, cobalt, and nickel on the fuel rod cladding surface will be activated and radioactive elements such as ^{54}Mn, ^{58}Co and ^{60}Co will be produced. These elements can be in various forms, depending on the coolant temperature, pH, electro-chemical potential, etc. Some of these elements may transport to the turbine system in direct cycle SCWRs and cause an increase in worker exposure. Fundamental coolant chemical properties as well as a better knowledge of the radiolysis of SCW are essential for this effort (this is discussed in more detail in Section 3.1.2). In particular we need to know the solubility of the radioactive elements in SCW over a range of temperatures (280-620°C).

Safety systems and safety related:

- Criteria: Because some SCWR conditions are different from LWRs (e.g., no water level and no boiling transition at supercritical pressure, different cladding material, etc.), the safety criteria for both transients and normal operation should be confirmed/rationalized.
- Plant control logic [scram, ECCS, RHR, containment, passive systems, etc.]: Same as above.
- Coolability: To minimize the design margins for core cooling under normal as well as abnormal conditions, it is desirable to perform necessary heat transfer test and thus to develop accurate and simple heat transfer models for SCW which incorporate geometrical heat transfer enhancing effects and flow characteristics including cross flow, stability and natural convection (this is discussed in more detail in Section 3.1.3).
- Severe accident mitigation systems: Most mitigation systems being developed for LWRs are applicable to SCWRs, however, design optimization of such systems is essential. Accurate information on hydrogen generation rates due to high-temperature water-metal reactions would facilitate the design of such systems.

Balance of plant:

- Turbine configuration: Because the steam enthalpy for the SCWRs will be slightly lower than for commercial SCW fossil plants, design modifications are essential for optimization.
- Reheaters: Because reheating does not occur in SCWR cores, unlike fissile boiler plants, a new re-heater must be designed.
- Deaerator: Because SCW radiolysis has not been fully understood, there is little information on the amount of noncondensable gas in the coolant. This information is essential for the design of the deaerator.
- Start-up systems and their operational procedures: The applicability of the start-up systems for the SCW fossil plants has been investigated for SCWR plants. Designs and operational procedures for the start-up systems must be optimized/rationalized.

I&C:

- In-core sensors: Mainly because the thermo-hydraulic conditions for SCWRs are different than those for LWRs', special attention must be paid to the in-core sensor designs.
- Plant parameter control systems: Because the thermo-hydraulic conditions, as well as the plant responses, for SCWRs are different from current LWRs, the plant parameter (pressure, temperature, flow rate, etc.) control systems must be optimized.

Building structures:

- Reactor building: Because of smaller reactor vessel per electricity generation for SCWRs, the reactor containment building will be smaller. It is essential to design a small reactor containment building with as much or more safety as the current LWR containment buildings.

Confirmation/achievement of Reliability under SCWR Conditions

Historically, the reliability of the components/systems for newly developed nuclear plants has been confirmed by tests under the various conditions (but not necessarily before their deployment). The confirmation tests might be substituted by small-scale tests or computer simulations due to recent computer technology improvements as well as experienced accumulated with the current LWRs.

Mechanical/structural integrity (some of this is covered in Section 3.1.2 above):

Target components

- Fuel assemblies
- Control rod guide tubes
- Core plate, fuel support (orifice)
- Nozzles
- Seal mechanisms, etc.

Conditions/evaluation

- Environmental qualification
- Irradiation, creep influences
- Seismic conditions
- Corrosion behavior
- Erosion behavior
- Flow induced vibration, fretting behavior
- Thermal fatigue (cycle), thermal stress
- Aging, etc.

System performance assurance and demonstration:

Target systems

- CRDMs
- Plant start-up systems
- Plant parameter control systems
- Main steam isolation valves
- Safety pressure relief system (safety relief valves, quenchers and suppression pool)
- Seal mechanisms, etc.

Conditions

- Normal conditions

- Transient conditions
- Seismic conditions, etc.

Safety related system responses at various initiating events:

Target systems

- Scram systems
- Reactor core isolation cooling (RCIC) systems
- ECCS
- RHR systems
- Safety pressure relief systems
- Depressurization systems
- Plant parameter control systems

Initiating events

- Normal shutdown
- Transient events
- DBA
- RIA
- Severe accidents

Verification of Software/Measurement Tools

The present software (design tools, evaluation tools, function logic) and measurement tools may lack appropriate models and have uncertainty because their applicable ranges do not always cover the SCWR conditions. Therefore, verification studies are essential for the following software as well as measurement tools.

- System design/evaluation tools (this is discussed in Section 3.1.3 above)
- Measurement tools under various conditions (e.g. level meter, flow meter)
- Sets of hardware/software logic for normal and safety functions (e.g., scram logic, ECCS logic).

Advanced O&M

The reliability of SCWR plants will be dominated not only by the design but also by the operation/maintenance. The following O&M items should be developed:

- Advanced O&M equipment [e.g. refueling machines, on-line maintenance systems, remote systems].
- Advanced O&M procedures [e.g. startup, shutdown, load following, refueling, fuel shuffling, inspection, maintenance].
- Verification of the plant/system operation [e.g. plant simulator].
- Mockup facilities for maintenance [e.g. coupling/uncoupling of control rods with drive mechanism, internals replacement].

3.1.6 SCWR R&D Summary Sheet

TWG-(1)
R&D Scope for the Supercritical Water-Cooled Reactor (SCWR) Concept
The low range of costs are for the thermal spectrum SCWR
Version (05-24-2002)

	Technical gap/issue				R&D items				
Sub-System	**Gap Label**	**Brief Description of Gap/Issue**	**Signific. of Gap *(a)***	**Current TRL *(b)***	**Activity Label**	**Brief Description of R&D Activity**	**Priority *(c)***	**Time *(d)***	**Estimated Cost Range (Million USD)**
Materials and Structures	M1	· Corrosion rates in supercritical water at temperatures between 280 and 620°C. The corrosion should be measured under a wide range of oxygen and hydrogen contents to reflect the extremes in dissolved gasses. · Composition and structure of the corrosion films as a function of temperature and dissolved gasses. · The effects of irradiation on the corrosion as a function of dose, temperature, and water chemistry. · Stress corrosion cracking as a function of temperature, dissolved gasses, and water chemistry. · The effects of irradiation on SCC as a function of dose, temperature, and water chemistry.	V	1	M1a	An extensive series of out-of-pile corrosion and SCC experiments on un-irradiated alloys	1	M	20
					M1b	Out-of-pile corrosion and SCC experiments on irradiated alloys	1	M	30
					M1c	In-pile loop corrosion and SCC tests	1	L to VL	180 to 250
					M1	Total			230 to 300
	M2	· The complete radiolysis mechanism in supercritical water as a function of temperature and fluid density. · The chemical potential of H2, O2, and various radicals in supercritical water over a range of temperatures (280-620°C). · Recombination rates of various radicals, H2, and O2 in supercritical water over a range of temperatures (280-620°C). · Effect of radiation (type – neutrons, gammas, as well as flux) on radiolysis yields. · Formation and reaction of other species by radiolytic processes.	V	1	M2a	Beam port and accelerator tests			
					M2b	Water chemistry control studies using the in-pile corrosion test loops			
					M2	Total	2	L	80
	M3	· Void nucleation and growth and the effect of He production on void stability and growth and He bubble nucleation and growth as a function of dose and temperature. · Development of the dislocation and precipitate microstructure and radiation-induced segregation as a function of dose and temperature. · Knowledge of growth or irradiation-induced distortion as a function of dose and temperature. · Knowledge of irradiation-induced stress relaxation as function of tension, stress, material, and dose.	V	1	M3a	Test specimen irradiation in the corrosion and SCC in-pile loops			
					M3b	Accelerator-based irradiation testing			
					M3c	Capsule irradiation tests in test reactors			
					M3d	Post-irradiation examination and analyses			
					M3	Total	1	L to VL	45
	M4	· Tensile properties as a function of dose and temperature. · Creep rates and creep rupture mechanisms as a function of stress, dose, and temperature. · Creep-fatigue as a function of loading frequency, dose, and temperature. · Time-dependence of plasticity and high temperature plasticity. · Fracture toughness as a function of irradiation temperature and dose. · Ductile-to-brittle-transition-temperature and helium embrittlement as a function of dose and irradiation temperature. · Changes in microstructure and mechanical properties following design basis accidents.	V	1	M4a	Un-irradiated alloy mechanical testing			
					M4b	Irradiated materials mechanical testing			
					Ma	Total	1	L to VL	100
						Materials total			455 to 525
Safety	S1	· Reduced uncertainty in supercritical water transport properties	V	1	S1	University or NIST out-of-pile studies	3	M	5
	S2	· Development of fuel to coolant heat-transfer correlations for SCWRs.	V	1	S2	Out-of-pile experiments starting with tubes and progressing to small and then relative large bundles of fuel rods.	2	M to L	15
	S3	· Supercritical water critical flow measurements	V	1	S3	Separate effects out-of-pile experiments	2	M	7
	S4	· Measurement of integral loss-of-coolant accident (LOCA) thermal-hydraulic phenomena in SCWRs and related computer code validation.	V	1	S4	Integral SCWR LOCA thermal-hydraulic experiments similar to the Semiscale experiments previously conducted at the INEEL	2	L	100 to 120
	S5	· Fuel rod cladding ballooning during LOCAs.	V	1	S5	Out-of-pile experiments using fuel rod simulators	3	M	5
	S6	· SCWR design optimization studies including investigations to establish the reliability of passive safety systems.	V	1	S6	Safety system design optimization and testing in existing out-of-pile facilities.	2	M	55
						Safety Total			187 to 207
Instability	I1	· The thermal-hydraulic, coupled neutronic/thermal-hydraulic, and buoyancy instability performance of SCWRs is not known. · The heat-transfer, pressure-drop, and critical-flow correlations are not validated for supercritical conditions.	P	2	I1	Out-of-pile instability experiments Verification of the the heat-transfer, pressure-drop, and critical-flow correlations Frequency-domain linear analysis of the onset of instabilities Instability total	2	M	12
Plant Design	P1	Design optimization and component performance and reliability assurance under SCWR neutronic and thermo-hydraulic conditions	O	3	P1	Design optimization and performance testing (where appropriate) of the fuel assemblies, control rod drive system, internals, reactor vessel, pressure relief values, coolant cleanup system, reactor control logic, turbine configuration, re-heaters, deaerator, start-up system and procedures, in-core sensors, containment building, etc.	3	L	120
Total cost									774 to 864

a Indicate relevance of technology gap: V = concept viability, P = performance, O = design optimization
b Indicate technical readiness level (1, 2, 3, 4, or 5); see EMG Final Screening Document
c Indicate priority of R&D activity:
1 = critical (needed to resolve a key feasibility or viability issue)
2 = essential (needed to reach a minimum targeted level of performance, or to resolve key technology or performance uncertainties)
3 = important (needed to enhance performance or resolve the choice between viable technical options)
d Indicate time required to perform R&D: S = short (<2y), M = medium (2-5y), L = long (5-10y), VL = very long (>10y)

3.2 Integral Primary System Reactors

3.2.1 Current State of Knowledge

The IPSR relies on proven LWR technologies, but in many cases in significantly different applications than existing reactor systems. For that reason there is minimal need for true viability research to support the IPSR, but there are significant development and engineering application needs.

Many design and R&D activities, including experimental work, have already been performed, in some cases for almost a decade, for the concepts comprising the IPSR group. They include:

IRIS:

- Completion of conceptual design.
- Testing of the steam generator performance.
- Testing and in-reactor operation of the steam generator on-line inspection and maintenance equipment.

CAREM:

- Completion of preliminary design.
- Completion of the basic design of a prototype reactor.
- Set up of experimental facilities.
- Performance of qualification testing of thermal-hydraulic, reactor control, and operating techniques in a High Pressure Natural Circulation Rig (CAPCN).
- Design and verification tests have been conducted successfully in a Cold Low Pressure Rig (CEM) for the internal CRDMs.
- Critical heat flux (CHF) tests were performed using a low pressure Freon rig and in water at high pressure and temperature.
- Tests were performed in a Critical Facility (RA-8) specifically constructed to support the CAREM project.

SMART:

- Completion of preliminary design.
- Design of small power prototype, with construction planned to start later in 2002.
- Set up of experimental facilities and start of the thermal hydraulic and integral component (steam generators, pumps, control elements drive mechanisms) performance testing.

MRX:

- Completion of preliminary design.
- Extensive testing, including materials lifetime testing, of the electro-magnetically driven internal CRDMs.
- Performance testing of some of the basic components.

This ongoing and rather advanced development status is a unique characteristic of the IPSR and has several important consequences:

- No major technology gaps exist. It is quite reasonable to expect that all the engineering gaps will be satisfactorily resolved.
- All the R&D required for the IPSRs can be successfully completed within this decade.
- The required R&D costs are quite limited, less than $100 million. Actually, the total estimated cost, including design certification, and up to start of construction for the IRIS first-of-a-kind (the highest priced of the IPSR group) is $460 million.
- The countries and organizations currently active in the IPSR development provide substantial cost sharing.

3.2.2 Gaps between What We Know and What We Need to Know

The gaps pertaining to the IPSR are “performance phase” gaps rather than “viability” gaps. There are two different kinds of gaps in relation to IPSRs, i.e., gaps that are necessary to be closed for initial deployment, and gaps necessary to be assessed for system optimization but not for deployment. Following is a list of the gaps belonging to the two categories.

Gaps to be closed for initial deployment:

1. Demonstrate operational characteristics and reliability of integral components.
2. Demonstrate safety characteristics.
3. Confirmation of integral reactor behavior.
4. Qualification of internal CRDMs.
5. Establishment of risk-informed licensing framework.
6. Development of on-line diagnostics and maintenance

Gaps allowing system optimization:

7. Development of long life, soluble boron free cores.
8. Optimization of multiple module arrangement.
9. Optimization of economics of multiple modules.

Gaps that are akin to design activities, which must be completed for a first-of-a-kind plant, have not been considered here, as these activities will be borne by the system proponent.

Since a description of the gap is intimately connected to how to resolve it and to its background, for convenience of the readers it has been decided to include in the following section both the rationale for, and the description of, the required R&D.

3.2.3 Proposed R&D Program

Following is the discussion of the why-and-how for the nine gaps (the first six for deployment, the following three for optimization) identified in the previous section.

3.2.3.1 Gap 1: Demonstration of Operational Characteristics and Reliability of Integral Components

The integral components (steam generators, pumps, pressurizer) experience different environmental conditions and thus ad hoc design, analysis, and testing is required. A discussion for each component follows:

Steam Generators

The IPSR adopts a relatively large number of helical, modular steam generators (e.g., 8 in IRIS, 12 in SMART and 12 in CAREM) to provide redundancy. Water/steam is inside the tubes, with the high-pressure primary coolant outside, thus the tubes operate in compression (and no tensile stress corrosion failures are possible). Operational data on integral steam generators exist and tests have been performed at Ansaldo, Italy, on a 20 MWth mockup of the IRIS steam generator. Major questions to be addressed are: failure modes under the integral reactor operating conditions, potential for parallel flow instabilities, and operational stability and control.

The required R&D is analysis, testing, and experimental verification of existing designs. Existing facilities may be utilized in Italy (Ansaldo), Russia (OKBM, a prospective IRIS Consortium member), Korea (SMART), and Argentina (CAREM). Estimated cost is $10 million.

Pumps

The low power reactors (e.g., MASLWR, the original CAREM version) have full natural circulation, and thus do not require pumps. Larger size reactors (IRIS, SMART, upgraded CAREM) do employ pumps. The most promising pump design is the IRIS fully immersed spool pump, an advanced version of the Westinghouse canned motor pump, which only requires a vessel penetration of the order of 2–3 in. This pump has been designed for chemical applications; it requires design and testing of key components (insulation, bearings) at reactor temperatures, and qualification for nuclear applications. Estimated necessary funds are $8 million.

Pressurizer

A variety of integral pressurizer methods have been adopted by the various IPSRs: heaters (IRIS); self-pressurization at low temperature (SMART), and self-pressurization at high temperature (CAREM). Steam/gas pressurizers are also possible. R&D is necessary to investigate advantages and disadvantages of various pressurizer concepts, followed by quantitative analyses of system transient responses for each pressurizer type, proof-of-principle and confirmatory testing and eventually design optimization of preferred concept. Estimated cost is $5 million.

Thus the total estimated R&D cost for this group (integral components) is $23 million.

It is expected that no major new facilities are required, and existing ones will be adequate for the proposed R&D with possibly a few exceptions (e.g., for the pressurizer testing) where relatively small ad hoc facilities may be required. The only materials development is for the spool pumps. Completion by 2006 is expected, assuming January 2003 as the starting date.

3.2.3.2 Gap 2: Demonstration of Safety Characteristics

The IPSR claims superb safety characteristics, which allow the elimination of traditional safety systems such as the ECCS. Among the IPSR group, IRIS is the concept that fully exploits the safety characteristics of the IPSR through its “safety by design” approach, while other designs like CAREM use a more traditional approach efficiently incorporating current safety features in a cost effective manner.

Small-to-medium LOCAs are shown to be without serious consequences through the thermal-hydraulic coupling of the vessel and the high design pressure spherical containment, which maintains core coverage throughout the transient. Testing will be necessary to confirm the analytical predictions, especially core coverage under a variety of transients.

The first R&D activity will be to perform a rigorous similitude analysis. While a full-scaled mockup is of course the preferred solution, in the vast majority of cases properly scaled models will be adopted. Thermal-hydraulic, as well as temporal similitude, will have to be demonstrated. This means that in some cases it will not be possible to simulate the entire transient, but scaled "segmented" models will be used by simulating only selected portions of the overall transient, with proper boundary conditions.

Many adequate testing facilities are available, such as those at Oregon State University and SIET, Italy, where the AP600 tests have been conducted or the PANDA facility at PSI, Switzerland, where the SBWR and SWR-1000 tests were conducted. Oregon State is currently conducting tests for AP1000 and MASLWR, a "trailer" of the IPSR set. Another attractive complex of test facilities is at OKBM, Russia where IPSRs have been designed and built. OKBM has a 200 MWth integral reactor and a series of test facilities that were used during its design. Finally, the SMART prototype is a very attractive choice when selecting the testing facility, however, it will not be available for several years.

It is expected that the tests in the out-of-pile facilities can be conducted over a 3-year period, followed possibly by a 2-year investigation (desired, but not required) of selected sequences in SMART.

Expected cost for the safety-testing program is $40 million.

3.2.3.3 Gap 3: Confirmation of Integral Reactor Behavior

The first gap (integral components) addresses the behavior of the individual integral components, however, interaction effects also need to be investigated (for example flow effects due to the positioning of the pumps on top of the steam generators, or the coupling of the pressurizer with the reactor vessel coolant). These effects will be investigated not only at steady state conditions, but also especially under abnormal conditions, for example, asymmetric behavior with one pump and/or one steam generator not operating or sloshing effects in the pressurizer due to a partially filled reactor vessel.

Testing facilities used for the second gap (safety characteristics) will be more than adequate to conduct testing related to this gap. Actually, it is expected that this series of tests will most probably precede the safety tests.

Expected required time is 2 years and expected cost of R&D is $4 million.

3.2.3.4 Gap 4: Qualification of Internal Control Rod Drive Mechanisms

The integral primary coolant system configuration is ideal for locating the CRDMs inside the reactor vessel. Internal CRDMs have several advantages: operational (there are no drive penetrations in the upper head, thus eliminating the head nozzles seal cracking and corrosion problems which have plagued the industry, with Davis-Besse being the last one); safety (no rod ejection accident is possible and no seal LOCA can occur); and economics (a more compact containment).

Most of the IPSR concepts thus consider internal CRDMs. The MRX has electro-magnetically driven mechanisms, which are quite similar to the conventional ones, with the difference being the in-vessel, rather than ex-vessel location of the motors. The other IPSRs have hydraulically driven CRDMs, where the hydraulic control system (essentially pumps and valves) is outside the vessel. An integral reactor (NHR5) with hydraulic CRDMs is currently operational at the University of Beijing in China.

Extensive testing of the electromagnetic CRDMs, including materials investigation, has been performed by JAERI and MHI for the MRX design. CAREM has performed extensive design and testing of the hydraulic CRDMs, and some preliminary investigations of the hydraulic CRDM have also been performed by the IRIS project. However, extrapolation from the small size NHR5, CAREM and MRX plants to plants with about 100 fuel assemblies is a major feasibility issue.

The required R&D includes two major areas. The first will address the reliability of operation and position indication under a variety of operating conditions. The objective is to have a robust, stable system that is responsive to demand, but is also insensitive to tolerances and variations in environmental conditions (e.g., coolant temperature). The second area is to design the hydraulic network or the electromagnetic drives inside the reactor to ensure ease of construction, operation, and refueling.

Facilities do exist where testing has already been conducted. However the construction of a specific high pressure and temperature facility, and the performance of endurance and qualification tests for the hydraulic drive system are necessary. Assuming that no additional materials investigations need to be conducted beyond the data obtained in Japan, the required R&D funding can be limited to $7 million and the required time to 3 years.

3.2.3.5 Gap 5: Establishment of Risk-Informed Licensing Framework

This is a United States issue and it is not a gap unique to the IPSRs. It is reported here because IRIS will probably be one of the first designs to go to U.S. NRC licensing under a risk-informed regulation framework. The IRIS approach to licensing is to combine defense in depth, represented by its safety by design, which eliminates most of the higher probability and consequence accident sequences, with a risk-informed approach that relies on probabilistic criteria to evaluate the remaining accident sequences. This will require development and application of an advanced PSA methodology, explicit treatment of uncertainties, development of a strategic process to develop risk-informed regulations, and finally design optimization using the risk-informed assessment methodology.

While the outcome of this process is of course directly applicable to IRIS, the methodology and the results of conducting this very process with the U.S. NRC will be of direct interest to all other advanced reactor designs that intend to follow risk-informed licensing.

It is expected that development of this risk-informed regulation licensing process will require 4 years at a cost of $2.5 million. This of course does not include cost of licensing, which will be borne by the specific project.

3.2.3.6 Gap 6: Development of On-Line Diagnostics and Maintenance

On-line diagnostics, important for all reactors, is particularly vital for the IPSRs, because of the integral configuration and long fuel life (5 years or longer in some designs). An R&D program is currently ongoing and will need to be expanded and focused. This will include, at least:

- Reactor core monitoring and control. Both in-core and ex-core detectors will be considered, but emphasis will be on in-core systems, because some IPSR, like IRIS, feature in-vessel shielding, which makes the vessel surface essentially non-radioactive. Westinghouse is currently pursuing the development of in-core SiC detectors. Reliable control rod position indicators are necessary, especially because of the adoption of internal CRDMs.
- Steam generator monitoring. Visual and ultrasonic in-service inspection systems have been developed and demonstrated by Ansaldo and Framatome for the Super-Phenix reactor. Current R&D is focused on developing eddy current, ultrasonic, and electro-magnetic acoustic transducers sensors

for detecting local degradation and deposit buildup. Methods to correlate on-line monitoring with a continuous estimate of time to failure are being investigated.

On-line monitoring and estimates of time to failure lead directly to the concept of preventive maintenance. For example, detection of excessive deposit buildup in the steam generators tubes will lead to cleaning of the affected tubes by chemical or mechanical means, which can be performed through the steam generator headers without opening the reactor vessel. This is just an example of the type of R&D necessary to develop a preventive maintenance program, i.e., a system where the IPSR systems are serviced before failure occurs, prolonging their useful lifetime. The possibility of performing such maintenance on-line at reduced power, taking advantage of the large redundancy in components, will be investigated.

Completion of the on-going R&D can be performed in 4 years at an estimated cost of $8 million.

3.2.3.7 Gap 7: Development of Long Life, Soluble Boron Free Cores

IPSRs feature a variety of fuel cycles, in terms of fuel (from less than 5% to slightly over 10% enriched UO_2, MOX, and U-ThO_2), cycle duration (standard refueling, 4-year fuel cycles, and up to a 10-years straight burn), and soluble boron (boron free, reduced concentration, or standard concentration). Burnable poisons are considered to provide high burnup straight burn capability and enable the soluble boron reduction. Advantages of straight burn and reduced or free boron cores are non-proliferation (less fuel accessibility), safety (more negative moderator coefficient and less re-fueling), economics (lower fuel cost, lower O&M costs, and systems simplification), and reduced waste (high burnup, possibly to 100,000 MWd/t).

While the related core design efforts have been conducted at length for all the IPSR designs, still a gap has to be closed to reach an acceptable solution for larger size cores with a straight burn cycle and no soluble boron. The required R&D will include investigation of various combinations of burnable poisons [e.g., Zr diboride (IFBA), Er, Gd] in various geometrical configurations; core management studies; control rod pattern studies; and assessment of radial and axial variable enrichment schemes.

These activities can be completed in 2 years at a cost of $2 million.

Development of high burnup fuels, either UO_2, MOX, or U-ThO_2, as well as more robust fuel such as metal dispersed or cermet fuels will also be very beneficial to the overall performance of the IPSR. However, it is not unique or a characteristic of the IPSRs, but it is regarded as a generic technology gap common to many reactor types.

3.2.3.8 Gap 8: Optimization of Multiple Module Arrangement

IPSRs are small-to-medium power modules, which are expected to be grouped together in power parks by those utilities requiring thousands of MW production. The gap described here (as well as the gap discussed next in Section 3.2.3.8) is not unique to the IPSRs and is shared with all small, modular plants. It is reported here because of the rather early projected deployment date for some of the IPSR designs. If, however, it is "allocated" to another concept, the IPSRs will be perfectly happy to share the results. The important issue is that this gap be addressed.

It is rather obvious that to be economically competitive, multiple modules in a single park will require sharing of essential functions including auxiliary systems, control rooms, and various plant site facilities. Conflicting requirements however do exist. For example, it is desirable to have the first module producing electricity while the second and third modules are still in construction; but this runs countercurrent to maximization of shared facilities. Another example is the adoption of a shared control room. It is the obvious solution, but consideration must be given to common mode features, human factors, and questions like what is the maximum number of plants that can be tied to a single control room.

The required R&D is the development of an analytical tool capable of accounting for design, fabrication, construction, and economic considerations to yield optimized arrangements and construction sequences for multiple module nuclear power plants. Design and licensing aspects (e.g., single control room) are not addressed, since they are not considered to be a bona fide gap.

Development of this tool is expected to require 2 years at a cost of $1 million.

3.2.3.9 Gap 9: Optimization of Economics of Multiple Modules

Small-to-medium power modules do not have the economics of scale of larger plants and therefore their economic potential must rely on different attributes. These are: design of simple components that are amenable to repeatable "mass production" fabrication, standardization, ease of transportation, and on-site assembly and erection.

Industrial fabrication methods have to be established and they should guide the design of the components. In the past, reactors have been designed generally as "one-of-a-kind" without much aforethought to subsequent fabrication and installation issues. Since the IPSR designs are well advanced and at the same time they are still at the stage when they can be easily modified, we have a unique opportunity to establish design procedures where multiple fabrication, transportation, and assembly considerations are integrated into the plant design. This will require collecting procedures from manufacturing industries and applying them to the design of nuclear components. Architecture-type design tools will also have to be developed.

An associated, very useful output of the resolution of this and the previous gap is that it will provide a realistic evaluation of the capital cost of small-to-medium nuclear power plant modules.

It is estimated that 3 years and $4 million will be required to perform this task.

3.2.4 Applicability of the IPSR R&D to Other Concepts

Most of the R&D tasks previously reported are of interest to other reactor concepts considered in this roadmap, as briefly discussed below.

- Resolution of Gap 3.3.3.2 (Demonstrate Safety Characteristics) will provide information that can be used to enhance the safety of other water reactors, e.g., the supercritical reactor.
- The integral configuration is not only characteristic of the IPSRs, but also of most liquid metal cooled reactors. Data obtained from 3.2.3.3 (Confirmation of Integral Reactor Behavior) will be of pertinent interest. Additionally, the IPSR testing facilities could be used for some liquid metal reactor simulation tests using a coolant that is cheaper and easier to handle than liquid metal.
- Because of the past and present operational failures of the vessel head penetrations, the results of the internal CRDM R&D (see 3.2.3.4) is of interest to all water cooled reactors.
- Development of on-line diagnostic and maintenance (3.2.3.6) is of general interest to all concepts. The technical solutions, such as in-core monitors, developed for the IPSRs because of their configuration, can be investigated by others to provide better performance and lower costs.
- Improvements in the economics of modular plants (3.2.3.7 and 3.2.3.8) are obviously applicable to all small-to-medium concepts, regardless of their coolant.

- Finally, the establishment of a risk-informed licensing framework (3.2.3.5) is applicable to all advanced concepts. IRIS will provide the initial experience for others to follow.

3.3 Pressure Tube Reactors (NG-CANDU)

This section describes the high-level R&D Program Plan to support the basic engineering design program for the development of a NG-CANDU nuclear power reactor. The R&D Program described in this document is currently underway in Canada.

The NG-CANDU is an advanced reactor design that is an evolutionary departure from the current CANDU designs. The NG-CANDU design is based on a change from heavy-water-cooled natural uranium fuel to light-water-cooled, slightly-enriched uranium fuel, while retaining heavy-water moderation and extending operating conditions to significantly higher temperatures and pressures. . The NG-CANDU requires significantly different components in several systems in order to accommodate these changes in fuel and coolant. As well, the NG-CANDU design must adopt improved systems and components, and improved design and construction methods to meet market requirements for a low capital cost product.

The technology gap addressed by the NG-CANDU R&D plan is primarily one of "Design Optimization", that is, pertaining to "performance" gaps rather than "viability" gaps. A Technology Readiness Level consisting of primarily the "Technology Development" and "Proof of Practicality" stages characterizes the level of development of the NG-CANDU design. The R&D program in many areas confirms that the impact of higher temperatures and pressures can be adequately predicted and accommodated. Other programs focus on the qualification of the design of several new components necessitated by some of the principal evolutionary changes introduced into NG-CANDU, e.g. the change to a smaller lattice pitch and SEU fuel.

The R&D program has five main goals:

1. Qualification of the fuel design.
2. Qualification of the fuel handling system design.
3. Completion of safety verification and validation activities.
4. Completion of component and equipment development and testing.
5. Completion of fuel channel design verification testing.

The elements of the R&D program can be divided into categories related to improvements in design (fuel, materials and components), safety, instrumentation and control, project delivery and O&M. The various technical sub-programs of the R&D program contribute in varying ways to these improvement categories. No major technology gaps exist and all engineering gaps will be satisfactorily resolved. All the R&D required for the NG-CANDU can be successfully completed within about 4 years. The required R&D costs are limited, less than U.S. $112 million. This number excludes the engineering work to bring this product to a "ready for sale state," licensing and advanced fuel cycle costs.

3.3.1 Fuel Program

The current NG-CANDU design will use a SEU oxide fuel in the proven CANFLEX fuel bundle configuration. The Generation IV reference NG-CANDU design will accommodate the use of extended burnup fuel and advanced fuel cycles that involve the dry recycling of spent LWR fuel (DUPIC). The Fuel program described below addresses the R&D for the SEU cycle only.

The fuel R&D program consists of five components: (1) fuel irradiation tests, (2) out-reactor fuel performance tests, (3) fuel thermal-hydraulic tests, (4) fuel design code validation, and (5) core physics and fuel management code validation.

3.3.1.1 Goal

Complete the fuel qualification program for the unique NG-CANDU fuel operating features listed below that are a departure from current CANDU conditions:

- Exposure to higher reactor coolant temperatures and pressures,
- Longer in-core residence under new thermal-hydraulic flow conditions, and
- Power ramping transients associated with refueling activities.

3.3.1.2 Current State of Knowledge

CANDU fuel has evolved from 7-element fuel bundles in the NPD reactor, through the 19-element fuel bundles in the Douglas Point reactor, the 28-element bundles in the Pickering reactors, to the current 37-element bundles in the CANDU 6 and Bruce and Darlington plants. The 43-element CANFLEX bundle is a logical extension in this evolution and has been demonstrated in CANDU 6 reactors.

CANDU 37-Element Fuel

The following summarizes some prominent features of the CANDU 37-element fuel bundle design that ensures low fuelling costs, good uranium utilization, high capacity factors and good fuel performance:

- High-density natural UO_2 pellets, which ensure dimensional stability. This ensures bundle dimensional compatibility with the fuel channel and fuel handling systems.
- Thin-walled collapsible Zircaloy-4 cladding for neutron economy and improved heat transfer. Improved heat transfer leads to low temperature and high fission gas retention within the pellets.
- No gas plenum. Extensive operating experience confirms that no plenum is necessary to accommodate fission gases, thus maximizing the fissile content per bundle. This experience includes extended burnup experiments and post-irradiation examination of commercial heavy water reactor fuel.
- High-integrity resistance welding of end caps, resulting in good fuel reliability.
- CANLUB graphite interlayer between the UO_2 pellets and Zircaloy cladding, which has eliminated fuel failures due to power ramping under normal operating conditions.
- Induction-brazed spacer pads that maintain separation of the fuel elements without the need of complex, expensive spacer grids.
- Simple bundle structure.

The in-reactor performance of CANDU fuel has been proven by the continuing successful operation in CANDU reactors. Of the 1,400,000 fuel bundles irradiated in Canada to 1996, more than 99.9% of the bundles have performed as designed. About half of the 0.1% defects can be attributed to a single-cause: SCC sheath failures caused by power boosts during the early refueling of Pickering Units 1 and 2 in 1972, and by overpowering the core of Unit 1 in 1998. Since the introduction of the graphite CANLUB sheath coating in 1973, there have been very few confirmed power boost defects during normal operations. Improved fuel management and adjuster rod sequencing, developed through operational experience, are also partly responsible for this reduction in defect rate.

The cumulative bundle defect rate for 37-element fuel in Canada is about 0.4%, of which less than half of these failures are attributed to fabrication and unknown causes. The 1997 annual bundle defect rate for CANDU 6 reactors is 0.073% in 25 000 bundles discharged from the 6 operating CANDU 6 reactors. About half of these bundle defects are attributed to debris fretting in newer units that have recently been placed in-service. Construction debris in the primary circuit occasionally is trapped within the fuel bundles and causes defects during the initial few years of operation.

CANDU fuel reliability and experience is comparable to PWR fuel experience. Furthermore, the operational implications of fuel defects are significantly less in CANDU reactors since on-power refueling and the on power failed fuel detection and location systems allow the removal of defected fuel without having to shut down the reactor.

NU 43-Element Fuel

AECL has been developing CANFLEX since 1986. Since 1991, AECL and the Korean Atomic Energy Research Institute (KAERI) have pursued a collaborative program to develop, verify, and prove the CANFLEX design. New Brunswick Power at the Point Lepreau Generating Station have recently completed a 2-year irradiation of 24 CANFLEX NU fuel bundles, as a final verification of the CANFLEX design in preparation for full-core conversion of a CANDU 6 operating plant.

The principal features of CANFLEX are enhanced thermal-hydraulic performance and a more balanced radial power distribution. The CHF enhancement appendages on the CANFLEX bundle enable a higher bundle power before CHF occurs, leading to a net gain in the critical channel power of 6% to 8% over the existing 37-element fuel. The maximum linear element rating in a CANFLEX bundle is 20% lower than that of a 37-element bundle. The lower element rating is achieved by adding extra elements and using larger diameter elements in the center rings, and smaller diameter ones in the outer two rings.

CHF experiments have been performed in Freon-134a in the MR-3 facility at the Chalk River Laboratories and light water at Stern Laboratories (Hamilton, Ontario) on both the 37-element and the CANFLEX NU simulated fuel strings for the CANDU 6 range of operating conditions. Also, the pressure drop characteristics of the CANFLEX bundle were determined in both Freon tests and hot and cold waters tests. KAERI tested a full string of CANFLEX bundles and 37-element bundles in their hot test loop at normal CANDU 6 operating conditions. AECL studied the axial pressure profiles for CANFLEX bundles in the Freon MR-3 facility.

Over the last several years, AECL and KAERI have subjected the CANFLEX fuel bundle to a set of out-reactor flow tests to simulate CANDU 6 reactor conditions and verify that the design is compatible with the existing C6 reactor hardware. In addition to the heat transfer and pressure drop tests (discussed above), the following mechanical flow tests have been successfully completed:

- Strength test: Fuel can withstand the hydraulic loads imposed during refueling.
- Impact test: Bundle can withstand bundle impact during refueling.
- Cross Flow: During refueling, when bundle is in the cross-flow region, the bundle withstands the flow-induced vibration for a minimum of 4 hours.
- Fueling machine compatibility: Bundle is dimensionally compatible with the C6 fuel handling system.
- Flow endurance: Bundle maintains structural integrity during operations – fretting wear on the bearing pads, inter-element spacers and pressure tubes remain within design limits over the 3000-hour test time.

CANFLEX bundles were irradiated in the U-1 and U-2 loops in the NRU research reactor to demonstrate performance under expected in-reactor conditions. Typical power changes during refueling and peak bundle powers during operation were calculated to establish the irradiation conditions for the NRU tests. Actual peak powers experienced were over 25% greater than in a CANDU 6. Detailed post-irradiation examination confirmed that all irradiation and design requirements were met.

CANFLEX NU bundles were also inserted into the ZED-2 facility at the Chalk River Laboratories to measure the fine-structure reaction rates, to validate the reactor physics lattice code WIMS-AECL. Reactor operation over 600 full-power days was simulated to determine peak bundle powers, power changes during refueling, burnup, and residence times. Various fuel schemes were studied. The CANFLEX bundle meets or exceeds all power requirements, and the data collected showed excellent agreement with code predictions.

The CANFLEX NU design was also analyzed for sheath strains, fission-gas pressure, end plate loading, thermal behavior, mechanical fretting, element bow, end-flux peaking and a range of other mechanical characteristics. Acceptance criteria established from years of operating experience with 37-element fuel and previous 37-element testing were met by the CANFLEX design.

The final step in the verification of the CANFLEX NU bundle is full-core implementation in a CANDU 6 core. CANFLEX has met or exceeded all design requirements in out-reactor tests and irradiation requirements from the NRU testing.

SEU 43-Element Fuel

To facilitate the achievement of extended burnup of interest in the near term (2-3 times NU burnup), AECL and KAERI have developed the SEU CANFLEX bundle.

The experimental irradiations in research reactors associated with the initial development of CANDU fuel all involved enriched uranium (in order to achieve sufficient ratings in the NRX and NRU research reactors). Some 66 bundles have been irradiated to high burnup (maximum 45-MWd/kg HE) at high powers, supplemented by data from the irradiation of 173 single elements. Over 3000 CANDU bundles have been irradiated to above average burnup in power reactors, with a few to a maximum of 30-MWd/kg HE. About 150 bundles have experienced burnup above 16.5-MWd/kg HE, mainly in (formerly) Ontario Hydro reactors. The extensive experience, both from research reactor and power reactors, demonstrates that current 37-element bundle design will operate successfully up to about 19 MWd/kg HE, at power and burnup levels representative of the 0.9% SEU/RU fuel cycles.

The lower element ratings with CANFLEX fuel will further increase confidence in good fuel performance at extended burnup.

DUPIC Fuel

AECL, KAERI, and the U.S. Department of State have collaborated since the early 1990s on an assessment of DUPIC. The CANFLEX bundle is the reference for the DUPIC cycle. Following a series of hot-cell experiments conducted at the Chalk River Laboratories to demonstrate fabrication of CANDU quality pellets using actual spent PWR fuel, a total of three DUPIC fuel elements were fabricated at AECL's Whiteshell Laboratories Shielded Facilities. Approximately 3 kg of spent PWR fuel was processed into fuel pellets using the OREOX process, and then the pellets were formed into stacks, loaded into three fuel elements, and welded. The DUPIC fuel elements were designed to mount on a special 37-element geometry bundle used for experimental irradiations in the NRU research reactor at Chalk River Laboratories. The irradiation of these elements in NRU started in the spring of 1999.

Measurements of the chemical content of the fuel before and after the DUPIC fuel fabrication process indicated that the volatile cesium, krypton, iodine, and xenon were released during the process. All other fission products and transuranic elements were retained in the fuel.

3.3.1.3 Gaps between What We Know and What We Need to Know

Fuel Irradiation

We need to confirm the irradiation performance of NG-CANDU fuel bundles to high burnup. The NG-CANDU fuel is an evolutionary development of the current CANDU fuel design. The CANFLEX fuel bundle has been qualified for use in CANDU reactors with natural uranium oxide fuel. CANDU fuel has been successfully irradiated in power reactors to burnups in excess of 20,000 MWd/tU. Nevertheless, the NG-CANDU fuel design represents an extension of the proven CANDU fuel performance database.

The current fuel design has a target burnup of 20,000 MWd/tU, with a maximum bundle power of 875 kW and a maximum linear element power rating of 45 kW/m. For the Generation IV timeframe, the target burnup is extended to 46,000 MWd/tU, with a concurrent evolution to the advanced DUPIC fuel cycle with a target burnup of 46,000 MWd/tU.

Fuel Performance Behavior

We need to demonstrate the irradiation performance of prototype NG-CANDU fuel bundles to confirm the performance limits for the NG-CANDU fuel during short-term power ramps.

During on-line refueling, the NG-CANDU fuel bundles will be exposed to power ramps as a result of fuel movement along the length of the fuel channel in the core and end-flux peaking at the terminal bundle in a fuel string within the core during the fuel movement. A series of irradiation tests will be undertaken to confirm the performance limits for the NG-CANDU fuel during short-term power ramps that are consistent with the unique NG-CANDU fuel operating conditions.

Fuel Endurance

We need to demonstrate that the NG-CANDU CANFLEX fuel bundle will meet all design requirements related to geometry, thermal-mechanical loads, and endurance, including:

- Cross-flow endurance in an end-fitting
- Vibration and endurance in a channel
- Refueling impact
- Compatibility with the fuel handling system
- Spacer interlock prevention
- Bearing-pad sliding wear.

Out-reactor tests that are representative of the more severe NG-CANDU operating conditions are needed to confirm the fuel performance behavior.

Fuel Thermal-Hydraulic Performance

We need to confirm the pressure drop and CHF behavior and associated correlations for the NG-CANDU CANFLEX fuel bundle over the range of normal operating and accident conditions.

Reactor power operating margins are established based on the thermal-hydraulic performance of the fuel. This depends on the details of the fuel design, the fuel power rating and the coolant flow characteristics. The NUCIRC thermal-hydraulic code is used to predict the thermal-hydraulic behavior of the CANDU primary heat transport circuit. This code contains correlations that are dependent on the fuel design and operating parameters. While the NG-CANDU fuel use the CANFLEX fuel bundle design, the bundle incorporates minor enhancements to improve the CHF margin of the fuel and to mate the bundle with the fuel channel design and the fuel handling system. The Fuel Development program includes a number of tests to obtain the data required to establish validated correlations for use in the NUCIRC code for NG-CANDU.

Fuel Design Code Validation

We need to validate the fuel design codes to the NG-CANDU requirements and conditions. AECL has established a suite of design codes to assist in the development of new fuel designs, including higher burnup SEU fuel.

To ensure that the fuel is designed to high quality assurance standards, the fuel design codes have been assessed for their applicability to the NG-CANDU requirements and any extensions in the code capabilities have been identified. As part of AECL's comprehensive software quality assurance program, an extension to the validation basis for the fuel design codes will be completed. This will involve formal verification and validation of the applicability of the fuel design codes to the NG-CANDU requirements and conditions.

Physics and Fuel Management Code Validation

We need to validate WIMS predictions for both the reactivity and kinetics of the new NG- CANDU core design and also the reactivity change associated with coolant voiding in the event of a LOCA.

The primary tool used to evaluate the physics of the reactor core is WIMS-AECL. This code was originally developed to analyze the core of the Steam Generating Heavy Water Reactor at Winfrith that had a core design, which was substantially equivalent to the NG- CANDU core design (heavy water moderator and light water coolant). The subsequent development and validation of WIMS-AECL has focused on the natural uranium CANDU core, and there is a requirement to extend the validation of the modern code version to the new design. The primary tool that will be used in the WIMS-AECL validation is the zero-power critical assembly at CRL, the ZED-2 reactor.

3.3.1.4 Proposed R&D Program

Costs for this program would be on the order of U.S. $20 million over 4 years and will be undertaken principally in AECL's Chalk River Laboratories (Chalk River, Ontario) and CANDU Products & Field Services Laboratory (Mississauga, Ontario).

The NG-CANDU SEU fuel development program activities are grouped in three categories:

CANFLEX NG Fuel Qualification

Fuel irradiation tests of the prototype NG-CANDU SEU fuel bundles will be conducted in the NRU research reactor to examine the effect of power ramps (during refueling), burnup (test target burnup of 25,000 MWd/tU), and coolant pressure (up to 15 MPa) on fuel performance parameters.

Full-scale 12-bundle string test will be undertaken in light water to confirm the pressure drop, CHF, and pressure drop behavior of the NG-CANDU SEU fuel bundles over the NG-CANDU range of conditions.

A series of in-reactor and out-reactor tests on prototype NG-CANDU fuel bundles and fuel elements will be undertaken to confirm the mechanical integrity, endurance, and vibration behavior over the NG-CANDU range of conditions. These test include:

- Sheath ridging autoclave tests
- Sheath corrosion autoclave tests
- Full-scale fuel string in fuel channel endurance tests
- Fuel verification separate effects tests
- Re-fuelling impact tests.

WIMS and Fuel Codes Validation

WIMS-AECL validation will require using the zero-power critical assembly at CRL, the ZED-2 reactor. Lattice arrangement experiments and lattice substitution experiments in this core will be used to validate WIMS predictions for the reactivity and kinetics of the NG-CANDU core design, the reactivity change associated with coolant voiding in the event of a LOCA, and to verify the reactivity properties of the reactivity control elements for the NG-CANDU. The work scope includes:

- Manufacture 280 driver fuel bundles for ZED-2
- Fabrication of 35 MOX substitution bundles in the Recycle Fuel Fabrication Laboratory at the Chalk River Laboratories for ZED-2
- ZED-2 tests for WIMS and shut-off rods.

Fuel code validation will be based on using the existing experimental database supplemented by the results of in-reactor and out-reactor tests on prototype NG-CANDU fuel bundles and fuel elements.

3.3.2 Safety

3.3.2.1 Goal

To establish the applicability of existing thermal-hydraulic and neutronic models, codes, and data for analyses of the NG-CANDU reactor.

3.3.2.2 Current State of Knowledge

Safety Code Validation

The NG-CANDU design will use the validated Industry Standard Toolset of safety analysis codes as the base technology to carry out the safety analyses required for reactor licensing. The IST codes are in the course of being fully validated for their application to the CANDU-type power reactors. The applicability of these codes to the analysis of the NG-CANDU reactor has been assessed. In most cases, the codes are directly applicable to the NG CANDU because of the similarity of the basic design.

High-Temperature Channel Behavior

The RD-14 safety thermal-hydraulic test loop has been upgraded to enable blowdown testing at the NG-CANDU heat transport system design temperatures and pressure, thus extending the validation basis of the CATHENA safety thermal-hydraulic code to cover NG-CANDU LOCAs.

Moderator Circulation Behavior

The smaller NG-CANDU core lattice pitch and the larger calandria diameter require a redesign of the moderator circuit for removal of core heat under both normal and abnormal conditions. AECL has validated the MODTURC code to predict the heat transfer for the fuel channels to the moderator using a ¼ scale moderator test facility for the current CANDU design. Changes to this facility will be implemented with the scaling required to validate the MODTURC code for application to the NG-CANDU core.

Severe Accident Behavior

The phenomena and the potential core behavior in the event of beyond design basis events in the NG-CANDU will be very similar to that anticipated for the current CANDU design.

Passive Containment Behavior

The NG-CANDU design will include passive heat removal capabilities.

3.3.2.3 Gaps between What We Know and What We Need to Know

We need to confirm the thermal-hydraulic and neutronic behavior of the NG-CANDU reactor under accident conditions including understanding the:

- Thermal-hydraulic and mechanical behavior of channels during high-temperature accident conditions.
- Thermal-hydraulic and mechanical behavior of the reactor core components (fuel channels, fuel bundles, etc) under severe accident conditions.
- Thermal-hydraulic behavior of the moderator circulation under normal and accident conditions.
- Thermal-hydraulic behavior of the passive heat removal systems.

3.3.2.4 Proposed R&D Program

Costs for this program would be on the order of U.S. $21 million over 4 years and will be undertaken principally in AECL's Chalk River Laboratories (Chalk River, Ontario) and CANDU Products & Field Services Laboratory (Mississauga, Ontario).

Fuel Channel Safety Assessments:

- Burst tests are planned for NG-CANDU conditions to demonstrate that the calandria tube surrounding a bursting pressure tube will not fail, as well as that no other fuel channels and no feeder pipes will fail.
- Continued assessment of a postulated single channel flow blockage/stagnation for NG-CANDU conditions is planned.
- Assessment of the behavior of the NG-CANDU fuel channel during a postulated large break LOCA is planned.
- Assessment of the behavior of the NG-CANDU fuel channel during a postulated combined LOCA and loss of emergency core cooling is planned.

Safety Codes Validation

In selected areas, the validation basis of the codes must be extended to cover the new range of application required by the NG-CANDU design and operating conditions.

Severe Accident Behavior

Studies are planned in a number of areas to confirm the predicted behavior. These include studies of the heat transfer rates for a fuel channel under loss-of-coolant and loss-of-emergency coolant injection.

Passive Containment Behavior

The details of these containment passive designs are still under review. When the design options for these systems have been finalized, tests will carried out, as required, in AECL's passive containment test facility.

3.3.3 Materials and Components

The use of light water coolant, a higher coolant temperature and pressure and a tighter lattice pitch necessitates an evolution in the fuel channel design and components for the NG-CANDU reactor.

The elements of this facet of the R&D program address principally qualification of materials and components, and improvements in O&M technology.

3.3.3.1 Goal

Complete the qualification program for the fuel channel and reactor components to confirm that the new design features can achieve the design lifetime performance targets.

3.3.3.2 Gaps between What We Know and What We Need to Know

The background and rationale for the qualification program of reactor components is summarized below:

Pressure Tubes

The central element of the NG-CANDU fuel channel continues to be a Zr-2.5%Nb pressure tube having a 103.4 mm minimum inside diameter for which AECL has generated a large knowledge base. The NG-CANDU coolant temperature and pressure will be 325^{o}C and 12.0 MPa respectively at the reactor outlet header. These conditions are slightly more demanding than the maximum coolant outlet parameters of 310^{o}C and 10 MPa for the current generation of CANDU reactors. To compensate for the more demanding operating conditions, the thickness of the NG-CANDU pressure tube has been increased from 4.2 to 6.5 mm. As a result, stresses within the tube are within the range of experience for the current generation of pressure tubes.

The general requirement for reliable operation of CANDU pressure tubes is that their deformation not exceed allowable limits, that their material properties remain acceptable throughout their design life, that they remain relatively free from flaws and that there be no crack growth mechanisms which could cause significant crack growth if a flaw that might initiate cracking exists. The lifetime for a CANDU pressure tube is primarily defined by two ageing phenomena:

- Deformation that will eventually exceed the allowable limits, and,
- Corrosion by the slightly alkaline coolant that flows inside these tubes. (The hydrogen uptake associated with this corrosion could eventually result in pressure tubes becoming susceptible to delayed hydride cracking at operating conditions.)

To minimize the rates of these two ageing processes, the NG-CANDU pressure tube design will take advantage of the accumulated experience and knowledge obtained from pressure tubes removed from operating plants, as well as from the extensive materials R&D program that has been the basis for evolutionary advances in CANDU pressure tube technology. In particular, the NG-CANDU pressure tubes will be fabricated using Zr-2.5%Nb with optimized microchemistry specifications and manufacturing processes.

The current state of knowledge about the two primary ageing phenomena for CANDU pressure tubes is summarized in the following:

Deformation. The hexagonal close packed crystal structure of zirconium results in an anisotropic deformation of operating pressure tubes that is primarily due to the preferential movement of point defects generated in the material by neutron irradiation. The deformation depends on the neutron flux, the material's microstructure, the crystallographic texture, the temperature, and the applied stress. As the structure of the material evolves with time due to irradiation, the deformation rate can change. Thus it is important for reactor designers to have a deformation equation that describes the functional relationships between strain rates and the conditions to which an operating pressure tube is subjected, so tube deformation can be predicted and then accommodated by providing appropriate allowances for the peak deformation that may occur at the end of its design life.

In addition to having established a deformation equation to predict the rate of creep and growth in its current pressure tubes, AECL has an R&D program in place to examine the behavior of thicker Zr-2.5% Nb pressure tubes at NG-CANDU operating conditions. This program will produce data using representative samples of NG-CANDU fuel channel material to extend the validation database for the deformation equation to the NG-CANDU conditions.

Corrosion (with associated hydrogen uptake). During reactor operation, the water flowing through pressure tubes slowly corrodes their inside surface and increases their oxide thickness. The loss of metal from this reaction is very small and does not limit pressure tube life; however the pressure tubes absorb some of the hydrogen produced by this corrosion so their hydrogen concentration slowly increases with time. The rate of this hydrogen uptake increases with temperature and thus is largest at the outlet end of the tubes. The main concern associated with an increasing hydrogen concentration in pressure tubes is when its solubility limit is exceeded at operating temperature. In such a case, hydrides could exist in the bulk of the pressure tube and could also be formed at flaws, if these are present. The former may reduce the tube's fracture toughness and the latter may result in delayed hydride cracking initiation, if a sufficiently large tensile stress exists at the flaw.

The fuel channel R&D program includes a number of experiments (both in-reactor and out-reactor) to extend the existing databases for pressure tube delayed hydride cracking and fracture properties, as well as the corrosion/uptake equation, to NG-CANDU conditions. This work is expected to confirm that the change to light water coolant has no effect on the corrosion/uptake process and that there is a reduction in the corrosion/uptake rate due to the use of optimized chemistry in the coolant and pressure tube specifications for NG-CANDU.

We need to confirm the pressure tube deformation process driven by a combination of thermal creep and radiation-induced creep under high flux and NG-CANDU temperature & pressure conditions. Specifically:

Confirm the deformation (sag, elongation, diameter increase and wall thinning) and corrosion (with associated hydrogen uptake) behavior of pressure tubes for NG-CANDU operating conditions.

Confirm the delayed hydride cracking and fracture properties of pressure tubes for NG-CANDU operating conditions.

End-Fitting Rolled Joint

In the CANDU reactor, the Zr-2.5%Nb pressure tube is connected to two Type 403 stainless steel end-fittings by means of a rolled joint. The rolled joint provides both a mechanical coupling and a leak-proof seal. To permit the tighter lattice pitch between pressure tubes that is a feature of the NG-CANDU design, the end fitting must be redesigned. This includes a new rolled joint to mate with the thicker pressure tube. AECL has had considerable experience with the rolling of joints between dissimilar materials, including the reference design concept for the NG-CANDU.

Corrosion and hydrogen uptake occur both over the bulk pressure tube inner surface and at the end of the pressure tube that is mechanically rolled into the end fitting. The temperature of the pressure tube at the rolled joint is much lower than the temperature of the bulk tube. There is an electrochemical coupling between the pressure tube and the end-fitting materials, and the tight fit between the materials in the joint region provides a site for crevice corrosion.

We need to confirm the material behavior of the roll-joint between the pressure tube and end fitting of the NG-CANDU fuel channel. Specifically:

- Confirm the crevice corrosion and the electrochemical coupling behavior for the rolled joint design under NG-CANDU conditions.
- Confirm the mechanical behavior of dissimilar materials at the end fitting rolled joint under NG-CANDU conditions.

Spacer

A new spacer to separate the pressure tube from the larger diameter calandria tube is being designed for NG-CANDU. We need to confirm the performance behavior of the new spacer design under NG-CANDU conditions.

Fuel Handling System

The NG-CANDU will include substantially new designs for fuel handing systems including: the fueling machine (including carriage), fresh fuel transfer, and spent fuel transfer. Changes to the designs of those systems in the CANDU 6 design are required to accommodate the higher temperatures and pressures of the NG-CANDU coolant system. In addition, the designs will be optimized to reduce capital and operating costs and to incorporate the feedback from the operating stations. The elimination of heavy water from the fuel handling systems offers opportunities for simplification and cost savings. We need to confirm the mechanical behavior of the integrated fuel handling equipment under NG-CANDU conditions.

Reactivity Control Devices

There are four sets of reactivity control devices in the NG-CANDU design: the Shut-Down System No. 1 rods, the zone control rods, the reactor regulating control rods, and the Shut-Down System No. 2 liquid poison injection system. The first three are solid mechanical devices that are vertically inserted into the core from the reactivity mechanism deck and controlled by electric motor drives, and the last is a group of tanks that inject liquid poison into the calandria water through a group of nozzles located above and below the reactor core.

The mechanical design of the first three sets of devices is quite similar. All are solid absorbing elements located within guide tubes. The SDS 1 system, in particular, will be very similar to the existing SDS 1 system in the CANDU 6 design. However, the shape of the solid absorber elements and guide tubes will be modified to mate with the tighter NG-CANDU lattice pitch. The SDS 2 design will be based on the proven CANDU 6 design.

We need to confirm the performance behavior of prototype mechanical reactivity control devices, including the electric motor drives and position units, in the tighter NG-CANDU lattice pitch.

Corrosion Behavior - Other components

The operating conditions for the NG-CANDU reactor are different from those of the current CANDU design, thus potentially affecting the material corrosion, corrosion product transport, and chemistry control issues associated with other NG-CANDU components.

We need to confirm the corrosion and hydrogen uptake rates, including the effects of radiolysis on the coolant chemistry, on prototypical NG-CANDU material under temperature and coolant chemistry controlled conditions representative of NG-CANDU conditions.

3.3.3.3 Proposed R&D Program

Costs for this program would be on the order of U.S. $55 million over 4 years and will be undertaken principally in AECL's Chalk River Laboratories (Chalk River, Ontario) and CANDU Products & Field Services Laboratory (Mississauga, Ontario).

Pressure Tubes

An R&D program is planned to validate the design of the fuel channel for the NG-CANDU reactor. As much of this work involves the pressure tube, it will be performed primarily using prototype NG-CANDU pressure tubes that are currently being manufactured. After these tubes are fabricated, their properties will be compared to those required by their Technical Specification and to additional guideline values. In addition, the following R&D work is planned.

Pressure Tube Deformation:

- Creep and growth specimens for 27% cold-worked pressure tube material, which is the work hardened condition for the current generation of pressure tubes, are to be irradiated in a high flux reactor at NG-CANDU conditions. Additional creep capsules will be tested to determine the thermal creep component of strain.
- To verify that a reduction in cold-work reduces the deformation rate at NG-CANDU pressure tube temperatures, additional experiments in a high flux reactor using 12% cold-worked pressure tube material are also planned.

Pressure Tube Corrosion and Hydrogen Uptake:

- Hydrogen uptake specimens are to be tested at NG-CANDU conditions in the corrosion test loop of a high flux reactor.
- Hydrogen uptake predictions for NG-CANDU conditions will be validated against loop test results.
- Comparative tests in both heavy and light water environments will be performed.
- Electrochemical tests will be performed for the temperatures and potential material couples to be associated with the fuel channel rolled-joints for NG-CANDU.
- Loop testing of prototype NG-CANDU rolled-joints is planned, including modifying an existing loop to run at NG-CANDU conditions and validating predictions against test results.

Pressure Tube Delayed Hydride Cracking and Fracture:

- Delayed hydride cracking initiation threshold and growth rate tests will be performed at NG-CANDU temperatures using irradiated pressure tube material.
- Fracture toughness tests will be performed at NG-CANDU temperatures using irradiated pressure tube material.
- Validation of the flaw assessment model for NG-CANDU conditions is planned.

Fuel Channel Components (End Fitting Rolled Joint, Spacer, etc)

- The required new fuel channel rolled-joint prototypes will be fabricated and tested.
- The required larger diameter calandria tube prototypes will be fabricated and tested.
- The required new rolled-joint between this calandria tube and the reactor structure prototypes will be fabricated and tested.
- The required new fuel channel spacer prototypes will be fabricated and tested.
- The required new fuel channel axial restraint prototypes will be fabricated and tested.
- The required new fuel channel annulus seal prototypes will be fabricated and tested.

Fuel Handling System

The qualification program of the integrated fuel handling equipment under NG-CANDU conditions and tighter lattice pitch constraints consists of the following sub-components qualification tests:

- Bore seal channel closure.
- Snout seal, ram and bundle separators.
- Shield plug/flow-through latched spacer.
- Fueling machine magazine.
- Fueling machine homing device.
- Integrated fueling machine testing.
- Spent fuel transfer mechanism.

Reactivity Control Devices

Qualification tests the SDS1 and SDS2 reactivity control devices will be undertaken in the tighter NG-CANDU lattice pitch configuration.

Corrosion Behavior - Other components

The materials corrosion program consists of three elements:

Corrosion Loop Tests:

- Design and construction of a corrosion loop covering NG-CANDU conditions.
- Corrosion tests (e.g., Flow-Assisted Corrosion, etc) of materials used in the Heat Transport System.
- Feeder material cracking tests.
- Core outlet chemistry verification tests.

In-Reactor (NRU) Tests:

- Corrosion tests (e.g., Flow-Assisted Corrosion, etc) of materials used in the Heat Transport System.
- Coolant chemistry verification tests.

Technical Specifications: An important element of improvements to O&M technology requires the development of chemistry and materials specifications

3.3.4 Information Technology

The area of control and instrumentation includes both the electronic hardware required to monitor and operate the plant, and the software and engineering tools that are used to design, construct, commission and operate the plant. The engineering tools are included in this area because of the very strong interaction between the hardware and software required to obtain, manage and interrogate the information required for effective plant operating and management. The elements of this facet of the R&D program address principally design improvements in instrumentation and control, project delivery improvements, and improvements in O&M technology.

3.3.4.1 Goal

- To develop an advanced control room design to reduce operator error and improve reliability.
- To implement an advanced suite of engineering tools that will be integrated with plant information systems to lower plant capital and operating costs.

3.3.4.2 Gaps between What We Know and What We Need to Know

The CANDU plants under construction in China contain major advancements in the areas of hardware and software technologies related to engineering tools, communication devices, and instrumentation and monitoring devices/processes. NG-CANDU will build on the China experience and address the following areas:

- Develop an advanced control center with smart diagnostics
- Develop communications technology to improve plant control and operability
- Develop improved integrated computer-aided tools for plant design, procurement, construction, and commissioning
- Develop methods for coolant chemistry monitoring and inspection technologies
- Develop system health monitoring capabilities.

3.3.4.3 Proposed R&D Program

Costs for this program would be on the order of U.S. $8 million over 4 years and will be undertaken principally in AECL's Chalk River Laboratories (Chalk River, Ontario) and CANDU Products & Field Services Laboratory (Mississauga, Ontario).

Instrumentation & Control - Advanced Control Center

The NG-CANDU design will incorporate advanced digital control instrumentation and data management/display systems. The R&D program will support the development of the control room design and verify the design's conformance to human factors engineering requirements.

An area that has been identified for significant cost reduction is the communications infrastructure of the plant. The NG-CANDU design will incorporate the use of broadband communications technology using fiber optic cables and wireless communications where appropriate to reduce the cost of the plant equipment and the time required for installation and communication.

Project Delivery

Within the context of the continuous evolution and improvement of AECL's engineering design and product delivery tools, the NG-CANDU program will support the adoption and implementation of specific tools and engineering processes that can be implemented on a priority basis to reduce the cost and schedule for a new NG-CANDU plant. An early priority, building on AECL's experience in using 3-D CADD tools for overall plant design, of the NG-CANDU program will be the adoption of 3-D CADD tools for mechanical component design, linked to advanced manufacturing methods. This will be used in the design and development of new fuel handling system components to reduce plant costs.

O&M Technology Improvement

Included in the advanced plant information systems, will be advanced system health monitoring capabilities. Implementation of these systems will build on AECL's work to improve the performance capabilities of the current CANDU plants, such as the development of the ChemAND system for plant chemistry monitoring. The NG-CANDU R&D program will support the implementation of these new capabilities in the control room design and field instrumentation.

3.3.5 Project Delivery – Constructability and Manufacturability

To achieve aggressive cost reduction and schedule targets, the NG-CANDU design will take advantage of advances in materials and manufacturing and construction technologies.

3.3.5.1 Goal

To establish materials specifications and to develop the application of particular construction technologies.

3.3.5.2 Gaps between What We Know and What We Need to Know

Advanced Construction Materials

The R&D program on constructability will capitalize on AECL's extensive experience in concrete technology that has been gained from the construction of CANDU projects, waste management programs, decommissioning projects, tunnel sealing experiments, plant life management studies, and refurbishment projects for containment and spent fuel bay structures.

We need to develop advanced construction materials and concrete technology to obtain cost and schedule savings and design improvement.

Modularization Technology

To reduce costs and schedule, the NG-CANDU plant will be constructed using a modularization approach. The development and implementation of modularization and pre-fabrication of systems and components will also capitalize on AECL's experience and recent joint constructability studies undertaken with companies such as Hitachi, Shimizu, and SNC.

We need to confirm and develop the module design and particularly the use of composite materials or structures as input to developing enhanced modular construction methods.

Engineering Tools

We need to develop engineering tools to facilitate extremely high quality of design and associated information, enhanced constructability and reduce construction schedule.

3.3.5.3 Proposed R&D Program

Costs for this program would be on the order of U.S. $8 million over 4 years and will be undertaken principally in AECL's Chalk River Laboratories (Chalk River, Ontario) and CANDU Products & Field Services Laboratory (Mississauga, Ontario).

Advanced Construction Materials

Areas of interest in materials and concrete technology include:

- Preventive and remedial measures for the potential loss of pre-stressing in containment structures
- Application of modern nondestructive examination testing methods for concrete structures
- Instrumentation and monitoring technology for containment integrity
- Alternative liner materials for spent fuel bays.
- Specification for advanced high-performance concrete to reduce the cost and schedule for reactor building construction.
- Low-porosity advanced concrete formulations will be examined for potential application as sealants in selected plant locations
- Alternative materials to replace steel liners on the reactor building base slab
- Alternative liner materials on reactor building perimeter walls.

Modularization Technology

Areas of interest in constructability technology include:

- Techniques for using prefabricated rebar assemblies
- Composite structure technologies and their implementation in internal structures, pre-fabricated permanent framework, and bridging systems
- Introduce passive cooling in the design of reactor building walls.

Engineering Tools

The continuing joint constructability studies focus on the use of advanced construction technology tools including:

- Fully integrated set of electronic tools for design, construction, procurement and operation,
- Construction management process for skids and modules, and,
- Interface requirements between 3D CADDS and construction scheduling tools.

3.4 High Conversion Water Reactors

The primary research needs for the HCABWRs concepts are in the areas of the core design including neutronics, thermal-hydraulics, and the development of new fuel cladding and core internals materials. There is also a need for improvements in the technologies used in fuel recycling (advanced aqueous or dry reprocessing and MOX fuel fabrication), which is an important aspect of these concepts.

The R&D done to date on this concept set has been primarily performed in Japan. The JAERI has lead a joint research program to develop reduced-moderation water reactors (RMWRs) in collaboration with the Japan Atomic Power Company and the Japanese reactor vendors (Hitachi, Toshiba, and MHI). This activity includes:

- Core designs with high conversion ratios and negative void reactivity coefficients
- Reactor system designs
- Thermal-hydraulic experiments and analyses to investigate the critical power and thermal-hydraulic characteristics of tight lattice cores
- Critical experiments to confirm the reactor physics characteristics
- Safety analyses of MOX fuel with high enrichment plutonium irradiated under a hard neutron spectrum
- Evaluation of MOX fuel reprocessing technologies for economical fuel cycle.

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 this year, will develop a reactor neutronics and thermal-hydraulics design based on a thorium fuel cycle.

3.4.1 Reactor physics

Goal

Negative void reactivity coefficients and conversion ratios more than 1.0 must be confirmed to verify the feasibility of the RMWRs. Therefore, a mockup experiment for the RMWR is planed in the fast critical facility at JAERI shown in Figure 9 to investigate the core characteristics and to estimate the calculation accuracy of the core design tools.

Figure 9. Cross-section of JAERI's fast critical facility.

Current State of Knowledge

JAERI's fast critical facility is designed for studying the physics characteristics of fast breeder reactor cores. Experiments are carried out to provide integral data for core design of a fast breeder reactor by building various simulating assemblies. The reactor assembly is divided into two halves, which are separated for loading, then brought together for operation. Experimental cores are built in the facility by hand-loading plates of reactor materials (uranium, plutonium, sodium, stainless steel, etc.) into drawers, which are then put in the desired pattern into each half of the assembly, a honeycomb of square tubes. The facility can accommodate large variations in fuel composition and core geometry.

Mockup experiments for a high conversion light water reactor (HCLWR) were performed in the 1980's in this facility. The neutron spectra of the mockup core for the HCLWR were close to those expected for the RMWR cores and the reactivity effects associated with moderator voiding were measured. The HCLWR mockup experiments were analyzed and the results showed some problems when the conventional calculation systems are applied to the RMWR design study. The results were also helpful for determining the core composition and measurement items to be used in the mockup experiment for the RMWR core.

Technical Gaps and R&D Requirements for Reduction of Gaps

Critical experiments performed so far in Europe and Japan were reviewed, but no useful data are directly applicable to the RMWR development. The mockup experiments for the HCLWR performed in 1980s were carried out with enriched uranium fuel or lower enriched plutonium fuel than planed for the RMWRs.

Experiments with plutonium fuel will start in JAERI's fast critical facility in 2002. A new group constant set will be generated from the newest evaluated nuclear data library JENDL-3.3 to improve the calculation accuracy. The total cost is estimated to be about $4 million for these experiments and the validation of codes used for the core design.

3.4.2 Thermal-hydraulics

Goal

A tight-lattice core is adopted in the RMWR concept to reduce the water fraction in the core. In general, the tight lattice core causes the CHF to be decreased and the pressure drop to be increased at the same mass velocity. A heterogeneous core arrangement with blanket assemblies and inner blankets is also adopted to attain a negative void reactivity coefficient. The heterogeneous core arrangement results in higher local power peaking in core. The core characteristics are different from those of previous thermal LWRs due to the different neutron energy spectrum. The change in the core response function may affect the flow stability in core.

There are several issues to be solved for the RMWR thermal hydraulics, as listed below:

- CHF in tight-lattice core,
- Core cooling during abnormal transients and accidents, and
- Flow stability during operation.

3.4.2.1 Critical Heat Flux in a Tight-Lattice Core

Current State of Knowledge

The CHF is a major concern for the thermal design of the RMWRs. Fuel rod to rod gap sizes between 1.0 and 1.3 mm in a triangular arrangement are proposed for the current RMWR designs. Low mass flow rates and a short core are also proposed to attain a high core void fraction and a negative void reactivity coefficient in the core. A stepwise axial power profile resulting from the inner blankets needs to be considered for the feasibility of particular designs. Because these thermal hydraulic features are quite different from those of conventional LWRs, it is very important to check the applicability of existing CHF correlations to the RMWR tight lattice core.

CHF tests were performed at JAERI using a 7-rod test section with a gap width of 1.0 mm and typical BWR operating conditions in order to check the CHF correlations used in the core design calculation. Figure 10 shows an example of comparisons between the measured and calculated results. The CHF calculated with the design correlation is lower than measured value under typical RMWR operational conditions. Toshiba also performed CHF tests using a 7-rod test section with a gap width of 1.3 mm under typical HCBWR operation conditions. The Toshiba results are similar to the JAERI test results. Toshiba checked the effect of the bowing of the heater rods on the CHF. The test results showed little penalty on the CHF due to bowing of the heater rods.

These results suggest that the current RMWR designs with a tight lattice core are reasonable and feasible although their configurations are quite different from the conventional LWRs.

Technical Gaps and R&D Requirements for Reduction of Gaps

Although several CHF tests were performed using chopped cosine or flat power profiles under various thermal hydraulic conditions, no CHF test was performed with a stepwise axial power profile simulating the inner blanket. Also, few tests simulating abnormal operational transients have been performed. Although the RMWR fuel assembly has more than 100 fuel rods, the scaling effects on the CHF are not clear at present.

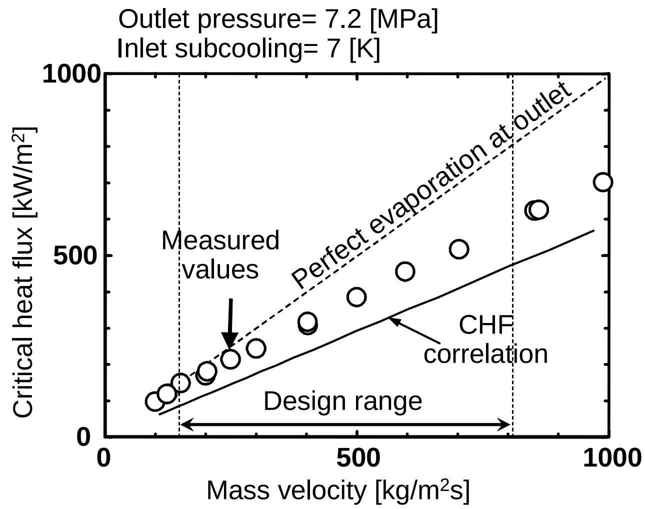

Figure 10. Comparison of measured and calculated CHF results.

The following R&D tests are required for the reduction of the technology gaps mentioned above:

- CHF test with axial power profile simulating the inner blanket,
- CHF test with a wider core to check the scaling effects, and
- CHF test under transient conditions.

It is also necessary to formally verify the thermal-hydraulic code so as to assure the reliable design of the RMWRs.

A CHF test with a 7-rod test section with axial power shapes simulating the inner blanket started at JAERI in April 2002 and will be completed by March 2003. Also, a 14-rod test section has already been fabricated at Toshiba to check the scaling effects. Toshiba plans both transient and stability tests as well as CHF tests under the normal operating conditions. In addition, a large-scale mock-up test is planned at JAERI for the proof of the RMWR design including transient tests, as shown in Figure 11.

The total cost is estimated to be about $20 million for these CHF tests and the validation of the thermal hydraulic codes used for the core thermal design.

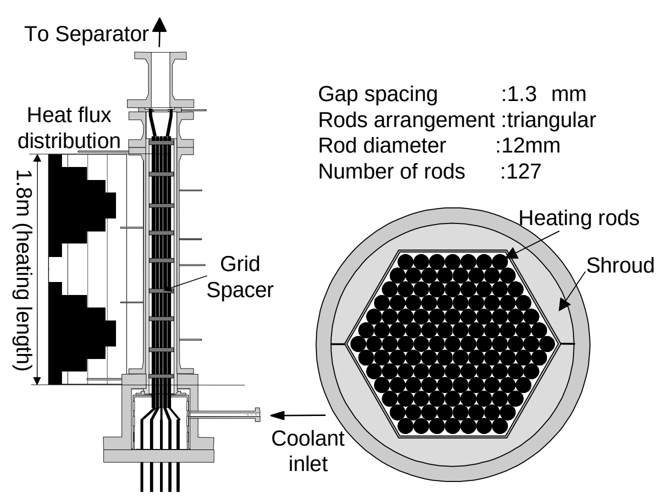

Figure 11. Schematics of test section for the full-scale mockup test.

3.4.2.2 Core Cooling during Abnormal Transients and Accidents

Current State of Knowledge

It is intended to use the reactor system of the existing plants as much as possible in developing the RMWRs. Therefore, the previous design, licensing, and operating experiences can be used for the RMWRs. It is expected that the current safety evaluation methods for abnormal transients and accidents for the ABWRs can be extended to the RMWRs by modifying several thermal-hydraulic correlations specific for RMWRs. The potential correlations that need to be assessed include the (a) core void fraction, (b) two-phase pressure drop through the core, (c) cross flow in the tight lattice core, (d) tie-plate counter-current flow limiting, etc.

Thermal-hydraulic feasibility studies are required to assess the RMWR core cooling performance during abnormal transients and accidents because of the RMWR design characteristics such as the tight-lattice core, use of an inner blanket, low core flow rate, high void fraction, lower void reactivity coefficient, etc. The thermal-hydraulic analyses performed at JAERI to date show that the RMWR cores are not susceptible to damage during typical DBAs, including pump seizure accidents.

Technical Gaps and R&D Requirements for Reduction of Gaps

As mentioned above, it is expected that current safety evaluation methods for abnormal transients and accidents for the ABWRs can be extended to the RMWRs by modifying several thermal-hydraulic correlations specific for RMWRs. The potential correlations need to be assessed include the (a) core void fraction, (b) two-phase pressure drop through core, (c) cross flow in tight lattice core, (d) tie-plate counter current flow limiting, and so on. It is necessary to establish test data for assessment of such correlations. These test data can be obtained using the same test facility as we are using for the CHF tests, therefore, there will be little additional facility costs. The total cost is estimated to be about $4 million for the assessment of thermal hydraulic codes.

3.4.3 Fuel design

Goal

The MOX fuels for the RMWR will contain more than 30% plutonium and will be irradiated to a burn-up of 100 GWd/tHM or over. These tough irradiation conditions make it an essential task to evaluate the mechanical and thermal feasibilities of the fuel.

Current Stage of Knowledge

JAERI conducted a preliminary safety evaluation of the fuel behavior using a fuel performance code called FEMAXI-RM. This computer code, which is an advanced version of FEMAXI-V, was developed for analysis of RMWR MOX and blanket fuel rods.

The analyses were conducted with for a single rod that is assumed to have the highest power in the RMWR core. The models or materials properties applied to the analysis, such as fuel thermal conductivity, fission product gas diffusion and release, and creep rate are derived or extrapolated from those used in the analysis of LWR fuel rods. The fission product gas release and rod internal pressure increase that is induced by the fuel temperature rise was of particular interest.

Figure 12 shows the fission product gas release rates from the MOX fuel. A sharp increase in the fission gas release in the MOX fuel appears at about 50 GWd/tHM, which is induced by a fuel temperature rise. However, after that, the fission gas release levels off due to the gradual decrease of the power. The fission gas release in the blanket part of the fuel is substantially null.

Figure 13 shows the fuel rod internal pressure rise, which is essentially generated by the fission product gas release. The pressure at end-of-life is less than 6.3 MPa, not exceeding the coolant pressure of 7.2 MPa. This predicts that the cladding will never cause "Lift-off," even at very high burn-ups.

Technical Gaps and R&D Requirements for Reduction of Gaps

The above analytical results suggest that the MOX fuel rod has no particular thermal behaviors that will raise safety and reliability concerns. However, the behavior of very high burn-up MOX fuel with such high plutonium content has not been fully researched or understood. Therefore, a more precise characterization of the input data and materials properties models is needed to evaluate the fuel safety and reliability on the basis of code predictions.

In addition, modeling of the degradation of the thermal conductivity with burn-up due to the swelling by the fission product gas pores which are generated around the plutonium-rich spots, and modeling of the fuel rod deformation behavior are main issues to be considered in the code analysis. For these issues, MOX fuel irradiation experiments are of vital importance. Also, further analyses is needed of the mechanical behavior of the fuel cladding, with an emphasis on pellet-cladding mechanical interactions (PCMIs) generated mainly by the pellet thermal dilatation and swelling.

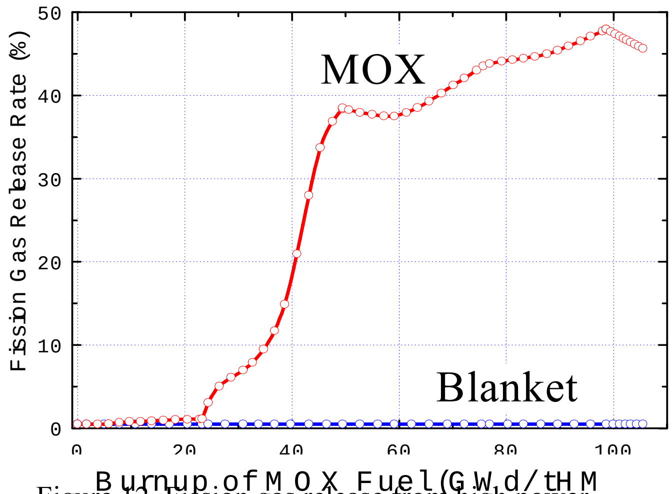

Figure 12. Fission gas release from high power MOX fuel.

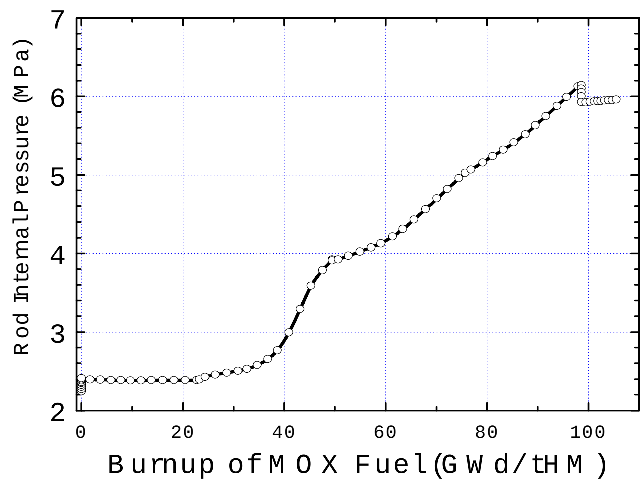

Figure 13. Fuel rod internal pressure as a function of burnup.

The total cost is estimated to be about $10 million for these MOX fuel analyses and irradiation experiments.

3.4.4 Cladding Materials

New cladding materials for use at ultra-high burnup (more than 100 GWd/tHM) and in a fast neutron spectrum may be needed. The R&D plan for the advanced reactor cladding will focus on acquiring data to confirm applicability and develop a mechanistic understanding of the material behavior during extended fast-spectrum irradiations. Nuclear properties (neutron absorption cross-section), irradiation damage, waste management (radioactivity), and the engineering feasibility for commercial production are all issues of interest. The key property needs include: compatibility with the BWR primary coolant, irradiation properties (dimensional and microstructural stability, mechanical strength, creep and ductility loss), and the resistance to pellet-cladding mechanical and chemical interactions.

Goal

The MOX fuel rod cladding material used in the ABWRs must be able to maintain integrity over burnups to 100GWd/tHM in both thermal and fast neutron spectrums. The important requirements are neutron economy, minimum irradiation damage, low radioactivity (minimal impact on the reactor maintenance and the waste management activities), and ease of commercial fabrication. A balance among these requirements will be needed for selecting candidate alloys and for confirming applicability.

Current Stage of Knowledge

Neutron Economy. The effects of the cladding on the neutron economy should be evaluated by considering several approaches for obtaining the needed mechanical strength. Even in the current BWR cores, the neutron spectrum in MOX fueled cores is harder than it is in UO_2 fueled cores. The advantages of zirconium alloys (with their low cross-section and neutron adsorption) may be partly offset because the wall thickness of the Fe-Cr-Ni base alloy cladding (with high mechanical strength) can be reduced by 1/2 to 2/3, compared to the zirconium alloys. Therefore, both zirconium alloys and Fe-Cr-Ni alloys need to be investigated.

Irradiation Damage. The cladding corrosion and irradiation damage depends on the neutron spectrum in each reactor core. The cladding corrosion and the permeation of hydrogen and oxygen into the cladding at the primary coolant side is controlled, in part, by the flux of high-energy particles, including the activated elements such as Co and Gd.

On the other hand, the metallurgical evolution of the cladding alloys due to the heavy neutron irradiation is controlled by the neutron spectrum. The important parameters that control the cladding irradiation damage include the total dpa and the He and P content formed by the transmutation reactions such as (n, α) and (n, p), respectively. The ratio of He/dpa and H/dpa is strongly dependent on the neutron energy spectrum. Some of the the stainless steels will face the problem of He induced void swelling. Triple ion beam irradiation using ion accelerators and neutron spectrum tailoring might provide the basic irradiation properties for understanding the microstructural evolution and irradiation-induced degradation.

Reactor Maintenance and Waste Management. The elements included in cladding alloys must be limited due to the formation of radioactive elements. Although the formation of short lived radioactive nuclei from zirconium alloys is lower than it is from Fe-Cr-Ni alloys, the radioactivity of the primary coolant circuit is controlled by the mass-transport of species included in the radioactive crud formed on the cladding surfaces and this is effected by the water chemistry. On the other hand, long-lived radioactive elements limit the spent fuel waste management and transport activities. These properties are strongly dependent on the cobalt content in the alloys. Fe-Cr-Ni alloys with low cobalt have post-irradiation radioactivities similar to the zirconium alloys. Water chemistry optimization will be required for fuel assemblies made of the new cladding alloys.

Feasibility of Producing Commercial Cladding Tubes. Technologies to produce economic cladding tubes will be required for the new cladding alloys. Based on the fabrication experience for making Type 304 austenitic stainless steel tubes, it should be possible to produce modified Fe-Cr-Ni base alloy cladding tubes. However, cost effective lining technologies for inhibiting PCMI may need to be developed.

Technical Gaps and R&D Requirements for Reduction of Gaps

The performance of the various potential new fuel-cladding materials during long irradiations should be evaluated by considering the degradation mechanisms expected in each material. The compatibility of the cladding with high temperature water will be affected by the primary coolant water chemistry. However, the irradiation effects due to the activated species of oxygen and hydrogen excited by low energy electrons formed under heavy neutron irradiation is an important factor for understanding the differences in the corrosion mechanisms among the candidate alloys. The susceptibility to IASCC will be influenced by the material degradation, the stress generated by the volume changes, and the water chemistry. The irradiation damage is strongly affected by the neutron energy spectrum and the transmutation reactions of the constituents of cladding alloys. In the case of Fe-Cr-Ni alloys, the accumulation of impurities and the depletion of the Cr along the grain boundaries caused by RIS is the most important factor on the susceptibility to IASCC. The potential for PC(M,C)I among the candidate alloys will depended on each candidate alloy.

Compatibility to the BWR Primary Coolant: Corrosion and Resistance to IASCC. The oxide growth rate of the zirconium-based cladding alloys is markedly accelerated in-pile compared with the out-of-pile data. The uptake of hydrogen and oxygen in a reactor is also accelerated and is dependent on the oxide growth rate. It is generally believed that this is due to the active oxygen and hydrogen that are formed by the excitation effects of the low energy electrons produced by the heavy neutron irradiation. The principle is similar to low energy plasma excitation. The corrosion acceleration of the zirconium alloys depends on the fast neutron flux and is essentially impossible to eliminate, because of the formation of ZrO_2 oxide films with N-type semiconductor properties. Therefore, zirconium-based alloy cladding cannot be used in ultra-high burnup fast neutron spectrum reactors.

Irradiation-induced acceleration effects on the corrosion of Fe-Cr-Ni alloys has been not observed, because of the formation of a double layer oxide film composed of P type M_3O_4 in the outer layer and N-type Cr_2O_3 in the inner layer. Also, the corrosion resistance of these alloys in high temperature steam (expected during a LOCA) may be improved by enriching the chromium content by up to 25%. Also, the degradation due to the formation of hydrides experienced in zirconium alloys is not observed in Fe-Cr-Ni alloys because of their low hydrogen solubility at normal irradiation temperatures.

The most important problem of these alloys is their susceptibility to IASCC accompanied with the segregation of impurities and the depletion of Cr along the grain boundaries. The RIS is strongly related to the austenite stability and the amount of super-saturated impurities at irradiation temperatures. Spinodal decomposition and martensitic transformation can occur at irradiation temperatures lower than 500°C in meta-stable austenitic steels such as Types 304 and 316. The degradation nose seen in the corrosion resistance and ductility is observed at that temperature region.

The resistance of Fe-Cr-Ni alloys to IASCC can be improved by combining the following technologies; enrichment of Cr (-25%) and Ni (-35%) to obtain sufficient austenite stability at irradiation temperature, purification of the austenite matrix by use of new melting methods such as electron beam melting, and a thermo-mechanical treatment (the so-called safety analysis report process - strained by heavy cold work, aged and recrystallized at intermediate temperatures). The alloy is modified into an annealed austenite matrix with fine dispersed precipitates. The mechanical strength of the ultra-low carbon alloy is maintained the same as for the Type 304 steels by the Hall-Petch effect, due to the fine grain structure.

A quantitative evaluation of the corrosion and environmental cracking of the candidate cladding alloys will be required to confirm the reliability of these materials. Also, the development of testing technologies for simulating the irradiation effects expected in cladding surfaces subjected to heavy neutron environments and heat transfer will be required along with post-irradiation tests in high temperature water.

Irradiation Properties: Dimensional Stability and Ductility Loss. Zirconium alloys experience significant irradiation growth and have relatively low creep strength under irradiation, both of which are dependent on the crystal texture. The reason is the strong crystal anisotropy that is characteristic of metals with an hcp type crystal structure (closed packed hexagonal structure). Therefore, texture control is required in the production of cladding tubes made of zirconium alloys. The irradiation growth is strongly dependent on the fast neutron flux. At high burnup, rearrangement of the crystal texture due to the internal stresses caused by the volume changes would be expected. Accordingly, the application of zirconium alloys to an ABWR operated with a fast neutron spectrum is difficult.

On the other hand, Fe-Cr-Ni alloys at temperatures lower than 500°C have excellent mechanical properties, as is well known from the fast breeder reactor experience. The important irradiation effects in the Fe-Cr-Ni alloys at ultra-high burnup are considered to be the ductility loss and channel fracture. Mechanistic analyses using new technologies would be required for a quantitative micro-structural evolution evaluation. The development of an indentation method with sensitivity at the nano to meso level is considered to be an effective means for identifying the grain-to-grain variation in mechanical properties. Also, quantitative irradiation creep data is required to establish an engineering database for optimizing the design of the cladding tube, fuel element, and fuel assembly.

Resistance to Pellet-Cladding Mechanical and Chemical Interactions. Zircaloy cladding in BWRs requires measures such a zirconium linier for inhibiting both PCMI and pellet-cladding chemical interactions. The effectiveness of a metallic zirconium coating will decrease with increasing the burnup. Moreover, the degradation of the inner surfaces of Zircaloy cladding at high burnup is expected to accelerate with an increasing accumulation of hydrogen and oxygen.

On the other hand, the pellet-cladding interaction countermeasures for Fe-Cr-Ni alloy cladding are easier than for Zircaloy cladding. The resistance to pellet-cladding chemical interactions is maintained by controlling the internal surface temperature. Therefore, the primary concern for these alloys is to inhibit PCMI and the release of tritium. The diffusion of tritium through the Fe-Cr-Ni alloys is higher than for the Zircaloys. However, it is possible to inhibit the tritium release by using a niobium base metal lining. Moreover, it is easier to maintain low fuel temperatures in fuel rods with Fe-Cr-Ni alloy cladding because they do not have significant outside surface oxide films.

However, it is necessary to accumulate empirical data under irradiation for identifying the effectiveness of niobium alloy linings for inhibiting pellet-cladding interaction and tritium release.

The total R&D cost for cladding materials is estimated to be about $10 million.

3.4.5 System design

Goal

The plant system must be designed for sufficient core cooling despite the use of a tight lattice core geometry, especially when the heat generation and stored energy in the core are relatively high. Since the insufficient cooling conditions may occur due to a rapid loss of core-flow or coolant, the RMWR systems have eliminated the large diameter piping for liquid-water flows.

Current State of Knowledge

The safety design for RMWR is based on well-matured technologies accumulated for the current generation LWRs especially for the ABWRs.

The reactor internal pumps installed in the reactor pressure vessel eliminate the need for re-circulation piping outside the vessel. The ECCS is a three-division system, with a high- and low-pressure injection pump and heat removal capability in each division functioning independently. One of the systems serves as the RCIC system, which has a steam driven high-pressure pump. The ECCS includes three on-site emergency diesel-generators to support the core cooling and heat removal if the off-site power is lost. The ECCS is designed to maintain core coverage for any postulated line break size during accidents.

Technical Gaps and R&D Requirements for Reduction of Gaps

Several accident management measures are planned for the mitigation of the effects of severe accidents, as is done for the ABWRs. In addition to those accident management measures, a PCCS may be utilized to prevent containment damage caused by over-pressurization due to the steam generation during LOCAs. Here the PCCS is a passive cooling system without relying on any pump operation. It is designed to have sufficient cooling capability for steam condensation with a conservatively estimated amount of noncondensables (nitrogen and/or hydrogen).

The PCCS heat exchanger is submerged in the water pool located outside the containment, and is connected to the drywell on the inlet side and the suppression chamber on the outlet side. This PCCS is characterized by the use of horizontal tubes for the heat exchanger. The use of a horizontal heat exchanger has several advantages over a vertical one, which includes the enhancement of its earthquake resistance, a reduction of the pool water level, and ease of maintenance. Also, the horizontal heat exchanger can be economically optimized, while optimization is not possible for the vertical heat exchangers because the pool liquid level limits the tube length.

The ROSA/LSTF facility at JAERI will be used to obtain the information useful for the safety confirmation and design optimization of this system. For example, large-scale tests are now being conducted to confirm the effectiveness of the horizontal PCCS, from which the promising results have been obtained. The total cost is estimated to be about $10 million for these tests.

3.4.6 Development of Reprocessing and MOX Fabrication Technology

Goal

The MOX reprocessing technology must be simplified to reduce the RMWR fuel cycle costs. One of the requirements is an improvement in the decontamination factors that will enable the application of the current MOX fabrication technology for the RMWR fuel production, in order to bring the RMWR into practical stage about 2020.

Current State of Knowledge

Spent MOX assemblies have the following characteristics compared with spent UO_2 assemblies:

1. Large plutonium content
2. Plutonium isotope vector shifts to heavier side
3. Large content of actinides, large alpha radiation, large neutron production rate, and large heat generation
4. Increased platinum content
5. Increased beta-emission nuclide (H-3) content and beta-gamma-emission nuclide (Ru/Rh-106, I-129) content.

The study of the dissolving conditions is important because the content of the nondissolving plutonium and nondissolving residuals possibly increases. The content of the nondissolving plutonium is less than 0.5% of the initial plutonium by improvement of MOX fuel fabrication process.

We now have significant experience in reprocessing fast breeder reactor MOX fuel assemblies. In Dounreay (England), 23t-HM of MOX was reprocessed and the plutonium dissolution ratio indicated that more than 99% was dissolved. In France, 28t-HM of MOX was reprocessed and the plutonium dissolution ratio was 99.8 to 99.9%. In Japan, fast breeder reactor fuel dissolving experiments have been performed and a simplified reprocessing technology with a high decontamination has been developed. Related to this reprocessing technology, the vibro-packing fuel fabrication method has also been studied. On the other hand, few reprocessing experiments for LWR-MOX have been reported. A mixing reprocessing method was reported in France in the 1980s. A mixed fuel composition that consists spent MOX fuel and spent / depleted uranium fuel is used for reprocessing. Recently COGEMA published information about a new mixing MOX reprocessing method.

Since the condition of the spent MOX fuel from the HCABWR (such as the exposure and plutonium content) will be situated between the LWR and the fast breeder reactor-spent fuel, the reprocessing technologies developed for LWR-MOX and fast breeder reactor-MOX are basically applicable to the HCLWR-MOX fuel reprocessing.

A technical evaluation was made of the single-cycle PUREX method, the pyrochemical method for oxide fuels, and the current PUREX method, in terms of the cost effectiveness, expected decontamination factors, and expected time until its practical use. The single cycle PUREX method is considered as the candidate technology for RMWR fuel reprocessing. A process flow was proposed based on the method, where the decontamination factors for fission product separation were expected to be about 10^5 (10^3 for Tc, 10^4 for Ru/Rh, and 10^7 for Am/Cm separation). In addition, Np, which adversely affects the core characteristics of the RMWRs, can be removed by the selective separation technology developed by JAERI with a decontamination factor of about 100.

Technology Gaps and R&D Requirements for Reduction of Gaps

Some modifications of the reprocessing facilities and operation process are required if the spent MOX fuel assemblies are reprocessed using the existing PUREX process facilities.

- The PUREX process consists of several steps (acceptance/storage, chopping and dissolving, separation, refining). The criticality management should be reinforced in some of the steps because of the increase in the plutonium content in the spent MOX assemblies.
- In the transport process, a transport cask that can withstand the increased heat and neutron loading should be used. Additionally, the cooling period before the transport should be extended.
- In the acceptance/storage processes, the worker radiation exposure during fuel discharge from the casks should be reduced. Additionally, the cooling capability should be increased.
- In the dissolving process, the dissolution speed must be optimized. There is the possibility that there will be more nondissolving plutonium because of the large plutonium content. The main nuclides of the nondissolving residuals are Ru, Rh, Pd, Mo, Tc. Since the yields of the Mo and Tc from the Pu-239 are large, the nondissolving residuals possibly increase.

Because it is expected that current reprocessing facilities can be used for the HCBWR MOX with some modification of both the facilities and operation process, little cost is required to develop the infrastructure. The total cost is estimated to be about $ 4 million.

4. RECOMMENDATIONS

The TWG1 recommended R&D activities for the water-cooled reactor system concept sets selected by the GIF for further consideration are presented in Section 3 of this report. In each case, it is the TWG1 view that all of identified activities will be necessary if that concept is to mature to the point of successful construction and operation of an actual plant. (We note that the costs and schedule of the individual tasks are in many cases difficult to forecast—but the needed R&D scope is relatively well understood.)

Of course, the actual formulation, scope, and implementation of the full Generation IV R&D Program will not be dictated by the TWG1 evaluations alone, and will depend on a variety of factors, such as:

- Final selection of concept sets for Generation IV R&D support, including water, gas, liquid metal, and nonclassical candidate
- Allocation of funding among these concept sets
- Timing of new reactor systems
- Emphasis and prioritization of work, determined on the basis of mission
- International participation.

The TWG1 therefore recommends that upon final selection of Generation IV concepts for Roadmap inclusion and R&D support a comprehensive and coherent multiyear R&D Program be formulated, taking into account the following considerations:

1. For each concept set, R&D work that is central to determining or confirming the viability of the concept should be identified and receive the highest priority; further, that work should be carried to the point that there is sufficient confidence in concept set viability before the supplementary R&D work (such as that needed to optimize design selections or maximize plant performance) are undertaken.

   In Section 3 above, most of the R&D activities are classified as either "viability" or "performance" tasks, consistent with this approach.

2. The multi-year Generation IV R&D support should take maximum advantage of synergy among reactor concept sets, with higher priority assigned to those tasks that can benefit more than one concept set. We recommend focusing on the R&D needs that are common across multiple reactor and coolant types, and fuel cycles.
3. Similarly, timing factors, and particularly leader/successor relationships among sets, should be considered, such that maximum learning from earlier reactor concept development and deployment can be applied to the later ones. For example, the logical progression from supercritical thermal to supercritical fast applications provides inherent strength and can avoid unnecessary R&D on downstream applications.
4. The final set of selected concepts, and the attendant R&D Program, should provide sufficient backup for the more innovative developmental aspects, to better ensure success for the longer term.
5. Industry support and co-funding are important factors. Preference should be accorded to those activities with strong support and significant co-funding, both as a matter of cost-effectiveness and because of the likelihood of continued down-stream support and full commercialization. (In this respect, we note that the preponderance of international experience in water-cooled applications may argue for a logical emphasis on these systems in the R&D Program.)

6. For any activity that is funded, there must be sufficient funding available in total (that is, from all funding sources, private and public) to provide reasonable confidence in completeness and success. It is not recommended that limited funding be broadly distributed over a range of inadequately supported R&D projects; experience demonstrates that successful R&D requires focused effort and resources, with margin to accommodate unexpected problems.

7. The R&D Program should focus primarily on support for the maturing of the technology and the underlying science for the reactor and its fuel cycle (as opposed to more narrow design development details). The technology and underlying science should enable the design development and eventual deployment of reactor systems that build on the conceptual approaches illuminated by the Generation IV Program.

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