Appendix 2.0
Supercritical Water Reactor

Contents

A2.1 INTRODUCTION AND BACKGROUND

Supercritical water-cooled reactors (SCWRs) are promising advanced nuclear systems because of their high thermal efficiency (i.e., about 45% vs. about 33% efficiency for current light water reactors [LWRs]) and considerable plant simplification. SCWRs are basically LWRs operating at higher pressure and temperatures with a direct, once-through cycle. Operation above the critical pressure eliminates coolant boiling, so the coolant remains single-phase throughout the system. Thus, the need for recirculation and jet pumps, pressurizer, steam generators, and steam separators and dryers in current LWRs is eliminated. The main mission of the SCWR is generation of low-cost electricity. It is built upon two proven technologies, LWRs, which are the most commonly deployed power generating reactors in the world, and supercritical fossil-fired boilers, a large number of which are also in use around the world. The SCWR concept is being investigated by 32 organizations in 13 countries. General information about the SCWR concept and its technical challenges is widely available in the literature, and it will not be repeated here.

In FY 2005, the SCWR program was redirected. The current plan focuses on further assessment of SCWR viability but independent of a specific concept design. Due to the potential economic benefits as an efficient electricity generator, reliable tools need to be developed to assess the viability of a variety of potential SCWR designs. Also, materials research needs to be conducted to establish the optimal (from a materials point of view) operational parameter range for SCWR and assure selection of structural and cladding materials that will maintain reliable operation of a SCWR power plant for its design life.

A2.1.1 System Description

Current research and development (R&D) programs within Generation IV International Forum (GIF) organizations address two principal SCWR design concepts that differ in the approach to the reactor design: one utilizes a reactor pressure vessel and the other pressure tubes. From its conception, the U.S. program focused on the reactor pressure vessel concept because its roots are in the LWR technology common to all U.S. reactor vendors. Similarly, the R&D conducted in Japan, Korea and Europe is focused on the pressure vessel concept (the main difference between the pressure vessel concepts lies in the core layout and different moderators). Canada selected a pressure tube design for its SCWR as the logical evolution of CANDU-type reactors. The U.S. Generation-IV SCWR Program operates under the following general assumptions, which are consistent with the SCWR's focus on electricity generation at low capital and operating costs:

- Direct cycle
- Thermal spectrum
- Light-water coolant and moderator
- Low-enriched uranium oxide fuel
- Base load operation.

These general assumptions are essentially common to all SCWR systems in consideration by the GIF except for the moderator (the Canadian system utilizes heavy water and the Korean system uses solid moderator). The reference SCWR design developed under this program is a direct cycle system operating at 25.0 MPa with core inlet and outlet temperatures of 280 and 500°C, respectively (these parameters are typical for all SCWR concepts under consideration). The coolant density decreases from about 760 kg/m^3 at the core inlet to about 90 kg/m^3 at the core outlet. The inlet flow splits with about 10% of the inlet flow going down the space between the core barrel and the reactor pressure vessel (the downcomer) and about 90% of the inlet flow going to the plenum at the top of the rector pressure vessel to then flow downward through the core in special water rods to the inlet plenum. Here, it mixes with the feedwater from the downcomer and flows upward to remove the heat in the fuel channels. This strategy is employed to provide good moderation at the top of the core. The coolant is heated to about 500°C and delivered to the turbine. The components limiting the power rating of the SCWR are the turbine and the reactor pressure vessel. Additional details about the U.S. reference design developed under this program and Nuclear Energy Research Initiative (NERI) projects are presented in the addendum of this Appendix.

A2.1.2 Overall System Timeline

The GIF SCWR Steering Committee has generated a schedule for the demonstration of the SCWR concept that calls for the completion of all essential R&D by 2015 and construction of a small-size (≤150 MWt) prototype SCWR by 2020.

A2.2 RESEARCH AND DEVELOPMENT STRATEGY

A2.2.1 Objectives

The objective of this 10-year plan is to identify activities needed to assess the technical feasibility of the SCWR concept. This ten years period roughly corresponds to the duration of the viability assessment phase as envisaged in the Generation-IV Technology Roadmap. The Generation-IV Technology Roadmap defines the objective of the viability assessment phase as: “Basic concepts, technologies, and processes are proven out under relevant conditions, with all potential technical show-stoppers identified and resolved.” The endpoint products for this phase, and therefore the objectives, were defined as follows:

1. Preconceptual design of the entire system, with nominal interface requirements between subsystems and established pathways for disposal of all waste streams

2. Basic fuel cycle and energy conversion (if applicable) process flowsheets established through testing at appropriate scale

3. Cost analysis based on preconceptual design

4. Simplified PRA for the system

5. Definition of analytical tools

6. Preconceptual design and analysis of safety features

7. Simplified preliminary environmental impact statement for the system

8. Preliminary safeguards and physical protection strategy

9. Consultation(s) with regulatory agency on safety approach and framework issues.

These endpoints and viability phase decisions (also published in the Roadmap) define the strategy and the framework for the scope of the R&D that needs to be conducted. Items 5 and 6 are the endpoints that this revision of the research plan addresses. The other endpoints depend on a specific design, which will be selected in cooperation with GIF partners. From the perspective of the U.S. program, the most important endpoint product is the item 5. The U.S. program does not follow any specific SCWR concept, but, to be able to assess the viability of proposed SCWR systems, robust and analytical tools are needed.

The initial work on SCWR conducted in the United States and elsewhere resulted in establishment of a baseline design that was needed to conduct the viability studies. Issues such as safety, stability, and overall performance were addressed. The R&D results so far indicate that there are no major or irresolvable technological issues associated with safety and stability of such a system. Great progress was made in safety system concept development, balance of plant design, reactor vessel design, startup procedure development, and stability analyses. The work pointed to the need of further development of specific thermal-hydraulic databases for development and validation of analytical tools. It also indicated that core design optimization is needed to assure reliable operation of the reactor under normal operating conditions. Because of the SCWR program redirection in FY 2005, the primary R&D focus will be on materials research and methods development that will assure that the viability phase endpoints can be reached. SCWR viability will be addressed using the U.S. baseline concept design generated in 2003-2004 and concept work conducted by the GIF partners. Therefore, the program will interface and coordinate with the GIF partners.

A2.2.2 Scope

The SCWR research and development program scope is organized based on the issues identified in the Roadmap. Its three main elements[1] are:

1. System Design
2. Basic Thermal-Hydraulic Phenomena, Safety, Stability and Methods Development
3. Materials and Chemistry

The first R&D element addresses development of the SCWR design and establishes a baseline design as reference for further feasibility and performance evaluation. Good progress was made on this element during 2001-2004 through activities under NERI and Generation-IV programs. A pre-conceptual U.S. baseline design was developed in collaboration with the industry. The core design was based on the Japanese concept that includes water rods with coolant down-flow for moderation. The dimensions and layout of the reactor vessel and internals were also determined, and the containment and initial concept of safety systems were designed. The balance of plant was conceptually designed and analyzed, and potential startup procedures were developed.

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[1] This R&D organization aligns the U.S. SCWR program elements with the GIF R&D plan. The three R&D elements correspond to the initial three SCWR GIF projects established. The original scope of the U.S. Generation IV SCWR research and development program was organized in two main elements:

- System Design and Evaluation
  - Baseline design for the core and reactor coolant system
  - Basic thermal data for the SCWR
  - SCWR safety systems and containment design
  - SCWR stability analysis
  - SCWR balance of plant, control, and start-up
- Materials Issues and Requirements
  - Reactor pressure vessel
  - RPV internals
  - Pumps, piping, and valve
  - Power conversion system

However, because the program was recently redirected, research activities will be focused on development of methods and the knowledge base needed for viability assessment, that is, the second R&D element. Comparative analysis of various core designs will be made to evaluate the relative merits and shortcomings of each, and their potential to meet the Generation-IV goals. The objective is to converge on a design that can be jointly developed and eventually demonstrated in cooperation with other GIF countries.

The second R&D area addresses current, basic knowledge gaps in areas such as the thermal-hydraulic phenomena expected during normal operation and accidents, system performance under a variety of conditions, and analytical methods needed for safety and system performance assessment. In collaboration with GIF partners, necessary experiments will be conducted, databases will be developed and analytical models and codes assessed and improved were necessary. Codes will be validated against available and planned experimental data and benchmarked against other codes developed by the GIF partners or elsewhere.

The third R&D element focuses on materials issues. The key to demonstrating the viability of the SCWR design is to identify and develop materials that can assure safe and reliable operation for the temperatures and pressures identified for the reference concept. The materials research is organized as follows:

- Oxidation, corrosion, and stress corrosion cracking
- Radiolysis and water chemistry
- Strength, embrittlement, and creep resistance
- Dimensional and microstructural stability.

The highlights of the planned R&D are provided in Section A2.3.

A2.2.3 Viability Issues

The Generation-IV Technology Roadmap identified availability of reliable materials and safety as the key viability issues.

The identification of appropriate chemistry and materials for in-core and out-core components is one of the main challenges for the development of SCWR. Zirconium-based alloys, so pervasive in conventional water-cooled reactors, may not be a viable material without some sort of thermal and/or corrosion-resistant barrier. Based on the available data for other alloy classes, no single alloy has currently received enough study to unequivocally ensure its performance in an SCWR. Although there is considerable experience with fast reactors and supercritical water-cooled fossil-fueled plants (FFPs), there is little or no data on the in-flux behavior of these materials at the temperature and pressure conditions of interest. Another key area needing greater understanding is the chemistry of supercritical water. The marked change in the density of supercritical water (SCW) through the critical point is accompanied by dramatic changes in chemical properties. These chemistry changes are further exacerbated by in-core radiolysis. , Preliminary studies suggest radiolysis of SCW is markedly different from what would have been predicted from simplistic extrapolations of the behavior encountered in conventional water-cooled reactors.

SCWR system safety issues are in many aspects similar to LWR; however, the specific characteristics of SCWR such as operational temperature and pressure, supercritical fluid properties, large variation in core coolant density in the axial direction, and low coolant inventory in the reactor system will make the SCWR system response significantly different from a LWR in accident conditions.

Therefore, demonstration of adequate levels of SCWR safety and availability of reliable analytical tools are critical viability issues.

Additionally, due to large the enthalpy rise in the core, SCWR systems are much more sensitive to hot channel factors than LWRs. Proposed core designs must be carefully analyzed for these factors to assure that the potential for hot spots exceeding allowable operational material temperatures is eliminated. To address these issues, good understanding of these phenomena and analytical tools, such as subchannel codes or computational fluid dynamics (CFD) codes, must be developed.

A2.2.4 Research Interfaces

A2.2.4.1 Relationship to GIF R&D Projects

Several GIF members have expressed strong interest in the SCWR concept including Canada, European Commission, Japan, Korea, and the U.S. A GIF "System Research Plan for the Supercritical Water-Cooled Reactor" was prepared outlining R&D activities to be conducted by the partners with the final objective of constructing a prototype SCWR system by 2020. The SCWR GIF partners have formulated and initiated three projects[2]:

- Design and integration—Definition of a reference design(s)[3] that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance. This work will involve identification of an achievable outlet temperature based on materials, fuel performance, and linkages to proven steam cycles in SCW FFPs.

- Basic thermal-hydraulic phenomena, safety, stability, and methods development—Significant gaps exist in the heat transfer and critical flow database for the SCWR. Data at prototypical SCWR conditions are needed. The design-basis accidents for a SCWR will have some similarities with conventional water reactors, but the impact of different thermal-hydraulic behavior and large changes in properties around the critical point compared to water at lower temperatures and pressures will have to be better understood and described more fully.

- Materials and chemistry—Select key materials for use both in-core and out-of-core and for both the pressure tube and reactor pressure vessel (RPV) designs. A reference chemistry will also be sought, based in large part on materials compatibility and the radiolysis behavior, which needs to be more fully described.

All GIF SCWR partners conduct domestic R&D programs that address the main SCWR technical issues discussed above. It is expected that the scope of the three projects identified above will be finalized and agreement signed by mid 2005. The project agreements will assure that the work conducted by the partners is complementary and that duplication is avoided.

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2 At the time of the writing of this plan, the multilateral project agreements were not yet signed; however, several bilateral agreements are in place. Project Management Boards were formed for each of the projects and initiated its work. This plan will be updated as the international collaboration evolves and specifics are identified for the established projects.

3 Pressure tube and RPV designs are both possible development paths for an SCWR.

A2.2.4.2 University Collaborations

Several U.S. universities contribute to SCWR R&D through direct participation in the Generation-IV program or variety of NERI and International NERI (I-NERI) projects.

The key participating universities are:

- Massachusetts Institute of Technology – conducting work on stability analyses
- University of Michigan – materials research
- University of Wisconsin – materials research
- Notre Dame University – materials research and chemistry.
- Rensselaer Polytechnic Institute – thermal-hydraulic data base and analytical methods

A2.2.4.3 Industry Interactions

Under NERI and Generation IV programs, close collaboration with two industrial partners was established. Westinghouse (2001-2004) was involved in the reactor concept, containment design, and safety systems development. Burnes and Roe (2003-2004) provided support in balance of plant design, startup, and development of control procedures. Both industrial partners provided substantial contribution to the conceptual system design and safety.

Through interactions with GIF partners, the project interacted with Atomic Energy of Canada, Limited; Toshiba; and Framatome Advanced Nuclear Programs (ANP) to develop SCWR concepts. These organizations are active participants either in bilateral I-NERI projects or through GIF SCWR project agreements.

A2.2.4.4 I-NERI/NERI

There are two NERI projects launched in 2001 that include significant design activities: one is led by the Idaho National Laboratory (INL) and involves the Westinghouse Electric Company and the other is led by the University of Wisconsin at Madison and involves the Argonne National Laboratory. Both projects have produced significant modifications to the Japanese and European baseline designs. Three new I-NERI projects will be initiated in 2005, with Japan for materials, with Korea for thermal hydraulic database, and with Canada for thermal-hydraulic database and materials. In addition, collaboration with EC and the International Atomic Energy Agency (IAEA) was initiated for the thermal-hydraulic database.

A2.3 HIGHLIGHTS OF R&D

A2.3.1 System Design

This research and development element provides the pre-conceptual SCWR design needed for the viability assessment and guidance for materials, thermal-hydraulic and system research. In general, this task addresses baseline design, safety systems, control and startup, system and comparative analyses, basic thermal-hydraulic phenomena, safety, stability, and methods. The work during 2005-2013 will be focused on cooperation with GIF partners and identification of the most promising design.

A2.3.1.1 System and Comparative Analyses

The objective of this activity, in cooperation with GIF partners, is to converge on a design that can be jointly developed and eventually demonstrated in cooperation with other GIF countries. The analyses will include operational analyses, safety analyses, and economic assessment.

A2.3.1.2 Basic Thermal-hydraulic Phenomena, Safety, Stability and Methods

This R&D program element addresses current basic knowledge gaps in areas such as the thermal-hydraulic phenomena expected during normal operation and accidents, system performance under a variety of conditions, and analytical methods needed for safety and system performance assessment. In collaboration with GIF partners, the necessary experiments will be conducted, databases will be developed, and analytical models and codes will be assessed and improved where necessary. Codes will be validated against available and planned experimental data and benchmarked against other codes developed by the GIF partners or elsewhere.

A2.3.1.2.1 Basic Thermal Data for the SCWR

Because of the lack of phase change in the core, SCWRs cannot use design criteria based on the critical heat flux concept. The commonly accepted practice is to specify cladding temperature limits that must be met during different events. This makes it very important to predict the heat transfer coefficient to the supercritical water coolant with great accuracy. However, while considerable information exists on heat transfer to supercritical water in round tubes for fossil boilers, little is known about the effect of the geometry and fluid conditions typical of the SCWR core. Therefore, this project addresses the critical issue of measuring heat transfer to supercritical water at prototypical SCWR conditions and to develop the tools to predict the SCWR thermal behavior.

Both actual SCW and supercritical surrogate fluids (CO_2 and Freon) will be used in this task. Surrogate fluids are convenient because some existing facilities already use such fluids. These fluids, in general, have considerably lower critical pressure and temperature, thus affording significant cost and time savings in constructing and operating the experimental facilities. On the other hand, SCW provides a direct representation of the SCWR behavior without the need for scaling of the thermo-physical properties.

The experiments should cover the various heat transfer regimes expected during operation of the SCWR, including upflow, downflow, and horizontal forced convection at high and low mass fluxes; buoyancy assisted forced convection; pure free convection; and deteriorated heat transfer. Transition from one regime to another, including the depressurization to two-phase flow conditions of an initially supercritical fluid, should be correlated in terms of dimensionless groups facilitating the comparison among different fluids, geometries, and flow conditions. In addition, in a bundle geometry where a circumferential symmetry does not exist, provisions should be made to measure the azimuthal variation of the heat transfer coefficient. The effects of flow channel shape, grid spacers, and non-uniform heat flux should be quantified.

The surrogate fluid work will consist of the following elements:

- Current facilities are suitable for single-tube or single-rod experiments only. It will be necessary to upgrade existing CO_2 and Freon facilities or construct a new CO_2 facility to accommodate a rod-bundle test section.

- Measure the heat transfer coefficient in single-tube and single-rod experiments to establish a connection with the fossil boiler database.

- Measure the heat transfer coefficient in bundle experiments.

To simulate the SCWR core, the SCW heat transfer facility will have the following requirements: pressures up to 27 MPa, bulk water temperatures up to 550°C, surface heat fluxes of up to 1.5 MW/m^2 with various axial power shapes, and a test section with a bundle of heated rods of proper and variable geometry. There exists a SCW facility at the Framatome-ANP laboratories in Erlangen, Germany (see Figure A2.1) that was used for supercritical fossil boiler tube tests and could be used by GIF for single-tube experiments. This facility also has a sufficiently large power supply and a pump to accommodate a relatively large heated-rod bundle. However, the actual bundle test section would have to be built as part of this project. The SCW work will then consist of the following elements:

- Upgrade of the Erlangen facility (the U.S. will design and construct the bundle test section).

- Measure the heat transfer coefficient at prototypical SCWR flow and geometry conditions.

In parallel with the experimental work, interpretation of the experimental data, including scaling to account for different fluids, geometries, and flow conditions, will be performed. This work includes the development and validation of best-estimate heat transfer correlations and models to predict the heat transfer coefficient in the SCWR core.

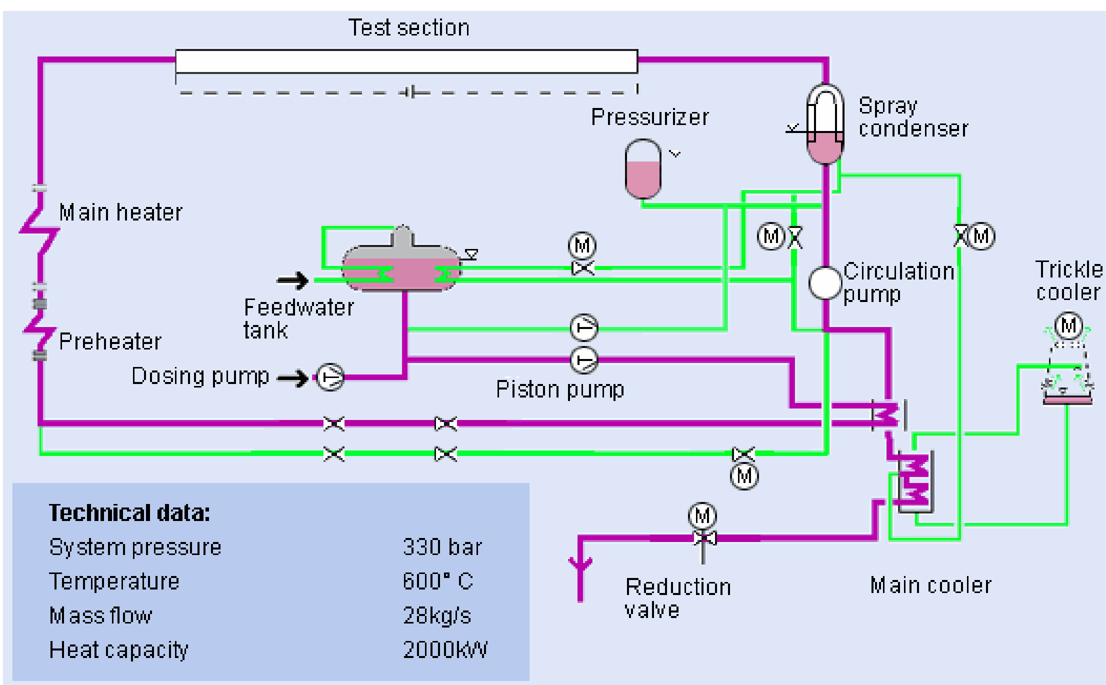


Figure A2.1. Flow diagram of the Benson test rig in Erlangen.

There is also a lack of data for critical (or choked) flow at supercritical conditions. Critical (or choked) flow phenomena are of great importance in designing/operating the reactor safety/relief valves and the automatic depressurization system as well as in the analysis of loss of coolant accident (LOCA) events.

A facility will be constructed consisting of a pressure tank, a discharge nozzle, various valves, measuring equipment, and data acquisition equipment. The design pressure and temperature will be <37 MPa and 500°C. The stagnation conditions in the tank as well as the diameter and length of the discharge nozzle will be systematically varied. Direct experimental measurements of the temperature, pressure along the discharge nozzle, void fraction, and flow rate at the nozzle outlet will be obtained. These data will enable accurate benchmarking of existing critical-flow models and the development of new ones.

The schematic diagram of a possible design for this test facility proposed by the University of Wisconsin at Madison is shown in Figure A2.2. The pressure vessel will be mounted to the ceiling and allowed to pivot on bearing assemblies to allow free movement opposite to the mass ejection. The momentum of the discharge will be measured by a force transducer on the side of the facility.

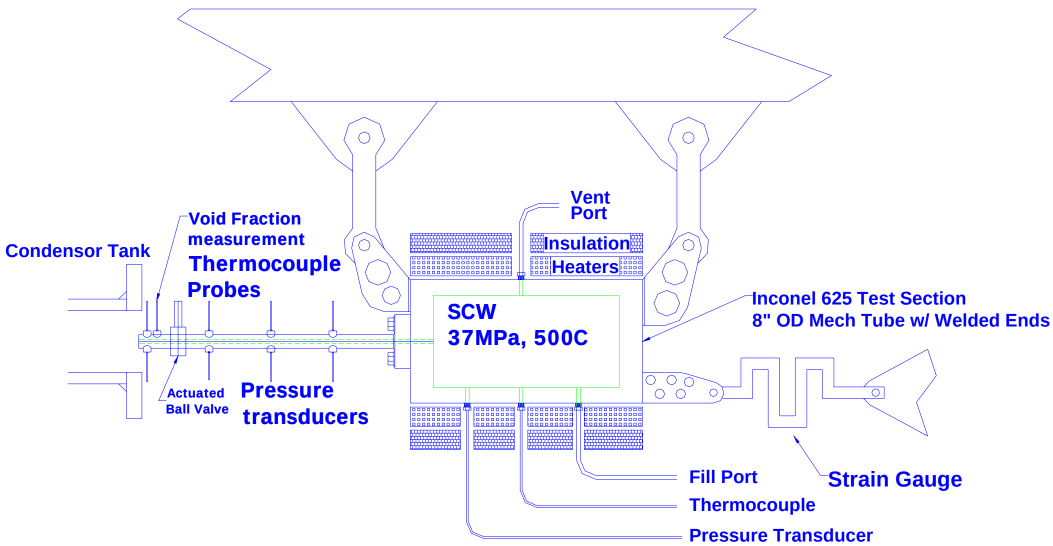


Figure A2.2. Supercritical water blow-down facility.

These basic data on heat transfer and critical flow will have to be rationalized and incorporated into existing computational codes such as RELAP, FLUENT, etc., for use in the SCWR analyses.

There is currently an internationally identified need/interest for an integral test facility that would provide data for code validation and demonstration of system performance under accident conditions. This facility would be an international project with participation by interested GIF parties and possibly non-GIF organizations. Such a facility can be designed and constructed only after significant progress has been made in the SCWR concept development, collection of basic thermal-hydraulic data from facilities, and methods development. In addition, since an integral facility will require significant funding, there must be sufficient participation and collaboration with other organizations. It is not expected that design and construction of such a facility could begin before 2009.

The development of the thermal hydraulic database will be an international effort conducted under GIF agreements and in cooperation with the IAEA, which will provide access to information beyond GIF partners. The Organization for Economic Cooperation and Development's Nuclear Energy Agency will provide guidance and support in development of the database.

A Phenomena Identification, Ranking, and Tabulation (PIRT) effort will guide and lead the data base development, as well as experimental planning, and model and methods development. The PIRT will be periodically updated as the database and experience with SCWR system behavior grows.

A2.3.1.2.2 SCWR Stability Analysis

SCWRs present the possibility of various types of instabilities, namely, density-wave instabilities, coupled thermal-hydraulic/neutronic instabilities, and natural circulation instabilities. It is necessary for any given design to show that either the oscillations do not occur during normal operation, or, if they do, that they can be detected and suppressed in a safe manner. Finally, oscillations under accident conditions must also be considered, e.g., under anticipated transient without scram conditions. The objective of this task 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 A2.3) identifying the stable operating conditions of the different SCWRs designs. Consistent with the U.S. Nuclear Regulatory Commission approach to boiling water reactor (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. 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.

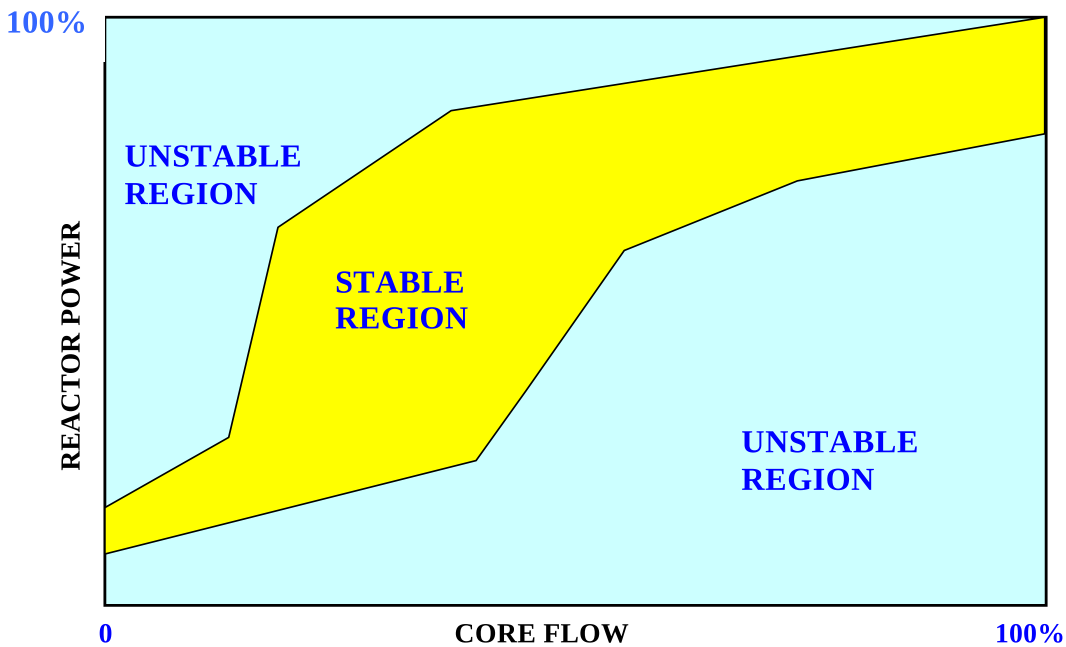


Figure A2.3. Conceptual Envelope for Stable Power-Flow Operation of SCWR

Therefore, in this task simplified analytical models will be developed to predict the onset of instabilities of the density-wave, coupled thermal-hydraulic/neutronic, and natural-circulation type. The models will capture 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 will also be assessed using analytical simulations. Parallel channel instabilities will be investigated as well as instabilities during start-up and partial load operation. Existing supercritical water and/or CO_2 loops will be used to perform experiments on both natural-circulation as well as density-wave type instabilities. These facilities will provide valuable data for benchmarking the analytical models.

A2.3.1.2.3 Methods Development

The objective of this R&D area is to assure that adequate and robust analytical methods will be available for evaluation of proposed SCWR concepts and their safety performance. The SCWR concept exhibits considerable similarities with current LWRs and pressurized heavy water reactors (PHWRs). They are all thermal systems that use UO_2 fuel, water coolant and moderator, and similar safety systems. The thermal-hydraulic and neutronic behaviors are coupled, albeit less so in pressurized water reactors (PWRs) and PHWRs, under steady-state conditions. Therefore, the neutronic and thermal-hydraulic codes developed for core and plant design, as well as analysis of steady-state, transient, and accident behavior of LWRs and PHWRs, are fundamentally adequate for the SCWR. However, three issues hinder reliable and accurate code performance for SCWR applications.

1. The large variation of the thermo-physical properties around the pseudo-critical temperature can cause numerical instabilities and execution failures.

2. The correlations and models implemented by the codes are not suitable to describe key SCWR thermal phenomena such as deterioration of heat transfer or critical flow at supercritical pressure. In addition, the codes are not validated for the tighter and more heterogeneous core geometries typical of SCWR designs with water rods[4]. Furthermore, a transition to two-phase flow conditions in the core has to be taken into account, which will happen in case of depressurization of the reactor pressure vessel, such as during normal shutdown or in emergency cases. The codes have to be capable to model such transitions.

With respect to neutronic analyses, there is a need to perform the following:

- Experiments at high temperatures (350°C to 600°C) to determine the impact of bound effects of hydrogen within water

- Experiments to investigate moderation conditions in the large water density range of interest to the Supercritical Water-cooled Reactor (0.1 to 0.8 g/cm^3).

The numerical robustness of existing codes will have to be tested with simulation of rapid depressurization and overheating transients. Based on the INL's experience with RELAP5, such transients can lead to execution failures. Examples of modifications needed to eliminate numerical instabilities include the use more of nodes as well as changing the interpolation expressions in calculating thermo-physical properties near the critical point. These approaches have been successfully applied to RELAP5 now used for safety analysis of SCWRs at the INL, Westinghouse, and Burns and Roe. However, further assessment against experimental data and code-to-code benchmark exercises will be needed.

Due to the high enthalpy rise in the core and the flow patterns in the reactor vessel plena, methods need to be developed that provide necessary the resolution on flow and temperature distributions beyond

[4]. A notable exception is the CANDU-X design, which retains the traditional CANDU pressure tube with circular geometry.

the capability of system analysis codes. This capability is needed to assess an optimized design and the safety of the SCWR reactor system. CFD codes are well suited to address these issues. Turbulence models are used in calculating the pressure distribution in channels and plena and in calculating heat transfer. CFD codes will need to be adapted for supercritical fluid conditions and assessed using experimental data. Coupling of CFD codes with system codes will be necessary to address properly a spectrum of transient and accident conditions.

To assure proper the basis for methods development and validation, an experimental database needs to be generated. Generation of this supporting database will be an internationally coordinated effort.

A2.3.2 Fuels

The SCWR concept is based on standard LWR fuel; therefore, there is no specific fuel research and development planned. The cladding issues are addressed in the materials research.

A2.3.3 Energy Conversion

The major components of the power conversion cycle are external to the reactor vessel and include the steam turbine and associated valving, the condenser, the demineralizer/condensate polisher, the feedwater preheaters, and the deaerator. There do not appear to be any special needs for alloy selection for the condenser, the demineralizer/condensate polisher, the feedwater preheaters, or the deaerator in the SCWR design, as long as the water chemistry guidelines developed for the control of corrosion in supercritical fossil plants can be followed. On the other hand, the turbine requires special consideration. However, initial studies and consultation with engineering and vendor firms have shown that the balance of plant and turbine issues can be resolved and are not a viability problem.

A2.3.4 Materials

This section describes in general terms the R&D needs for SCWR materials. The actual R&D needed to select and/or develop materials that meet these requirements is described in Appendix 9.0, Materials. Addendum 2 to this Appendix describes the materials requirements.

For any of the proposed SCWR designs, R&D on materials will need to focus on the following key areas:

- Oxidation, corrosion, and stress corrosion cracking
- Radiolysis and water chemistry
- Strength, embrittlement, and creep resistance
- Dimensional and microstructural stability.

In addition to these performance factors, the cost of the material and its effect on fuel utilization must also be considered to meet the economic and sustainability requirements of Generation IV designs.

For any SCWR core design, materials for reactor internals and fuel cladding will need to be evaluated and identified. Zirconium-based alloys, so pervasive in conventional water-cooled reactors, will not be a viable material for most of the proposed SCWR core designs without a thermal and/or corrosion-resistant barrier.

Based on the available data for other alloy classes, no alloy currently has received enough study to unequivocally ensure its viability in an SCWR. An extensive review of potential materials is not presented here but, based on experiences from LWRs, PHWRs, fast reactors and supercritical water-cooled FFPs, iron-based austenitic stainless steels (e.g., 304L, 316L), austenitics with higher Cr content (e.g., 690, 800), corrosion-resistant ferritics (e.g., HT-9), and advanced ferritic/martensitics (e.g., 9–14% Cr), should be given consideration as materials for both fuel cladding and core internal components. Precipitation-hardened Ni-based alloys (e.g., 718, 625) should also receive attention for applications where dose rates are on the lower end of the projected range. Thermal and corrosion-resistant barriers (e.g., stabilized ZrO_2) are another class of materials that could prove very useful in an SCWR environment.

There is little data on the general corrosion behavior of any of the candidate materials in supercritical water or at temperatures between current water reactors and the pseudo-critical temperature. Below the pseudo-critical temperature, the density of water is similar to a conventional liquid, and the chemistry is expected, therefore, to be similar to that in the liquid phase. Increases in temperature will enhance the general corrosion rate, but there is evidence that the failure mode may shift from wastage to stress corrosion cracking (SCC) in some materials. Above the critical point, the density of supercritical water is sufficiently low that the corrosion behavior is similar to a gas. Some studies have indicated that the corrosion rate may actually be highest just below the pseudo-critical point, which is consistent with a change in mechanism from corrosion by a high temperature liquid to a high temperature gas. The change in ionic solubility that occurs by heating through the critical point will also lead to different levels of impurities that will depend on the location within the circuit.

Failure modes such as SCC, pitting, inter-granular attack, inter-granular stress corrosion cracking, and irradiation-assisted stress corrosion cracking have been observed in many and varied components of conventional water-cooled reactors. The increase in temperature and change in water chemistry associated with SCW may exacerbate or arrest these modes in the candidate materials. Stress corrosion cracking has been observed in supercritical water in tests associated with the destruction of hazardous waste by the supercritical water oxidation (SCWO) process. The water chemistries that result from SCWO are typically much more severe than will be expected in the SCWR, but some insight on SCC and related failure mechanisms may be gleaned from these studies.

A key parameter in defining the corrosion behavior in an SCWR will be an enhanced understanding of the chemistry of supercritical water. The marked change in the density of SCW through the critical point noted above is accompanied by dramatic changes in chemical properties. For example, the ionization constant reduces from 10^{-14} to 10^{-23}, hydrogen bonding is greatly reduced or non-existent depending on the pressure, and the dielectric constant is reduced by more than an order of magnitude. These changes mean that the ionic solubility, pH, and corrosion potential will be distinctly different at the core inlet compared to the outlet. These complications are further exacerbated by in-core radiolysis. To understand the corrosion potential in an SCWR core and to effectively control corrosion rates, it will require knowledge of the corrosion potential of each of the stable and reactive-intermediate species present in-core. The radiolytic yields and recombination rates in SCW are currently unknown, and preliminary studies suggest a markedly different behavior at SCW conditions compared to what would have been predicted from simplistic extrapolations of the behavior encountered in conventional water-cooled reactors.

The viability of a SCWR will also depend on meeting challenges associated with the mechanical behavior of materials in- and out-core. Thermal and irradiation creep will be higher at SCW temperatures and pressures, although the enhancement in rate between those observed in conventional water-cooled reactors and conditions equivalent to an SCWR will depend on the material. The production of second phases, segregation, and/or the generation of He within a given material irradiated at high temperatures can lead to sufficient embrittlement that its integrity may be questionable during (and/or following) an outage. In addition, fatigue and thermal ageing effects on toughness are not well established for most of the candidate materials at SCW temperatures.

It will also be important to determine the irradiation-induced changes to the cladding and structural materials due to growth, swelling, helium-bubble formation, dislocation microstructure, precipitate microstructure and irradiation-induced composition changes, and that these changes will not compromise the integrity of the components for the design life of the reactor. Although focusing on the thermal core will reduce the magnitude of some aspects of the radiation-induced changes due to the lower fluence, He segregation will be an important consideration because of the greater relative production of He/dpa at thermal neutron energies. For temperatures between 280 and 350°C, the irradiation damage behavior for most of the materials under consideration is fairly well known. Swelling due to void formation and irradiation-induced segregation can be modeled adequately for all except the ferritic/martensitics where the data is more limited. For temperatures between 350 and 625°C, all microstructure features change quickly with temperature. At the lower end of the range, He segregation and precipitation may lead to enhanced grain boundary embrittlement whereas at the higher end the microstructure of most materials will resemble the annealed condition because of fewer vacancy loops and a lower network-dislocation density. Most of the available data on the high-temperature irradiation effects are from fast reactors with a higher fast-to-thermal-flux ratio. Some data is available from mixed-core reactors, but the R&D program will have to examine the behavior at SCWR conditions to ensure that mechanisms and rates are equivalent.

A2.4 10-YR PROJECT COST AND SCHEDULE

A2.4.1 10-yr Project Budget

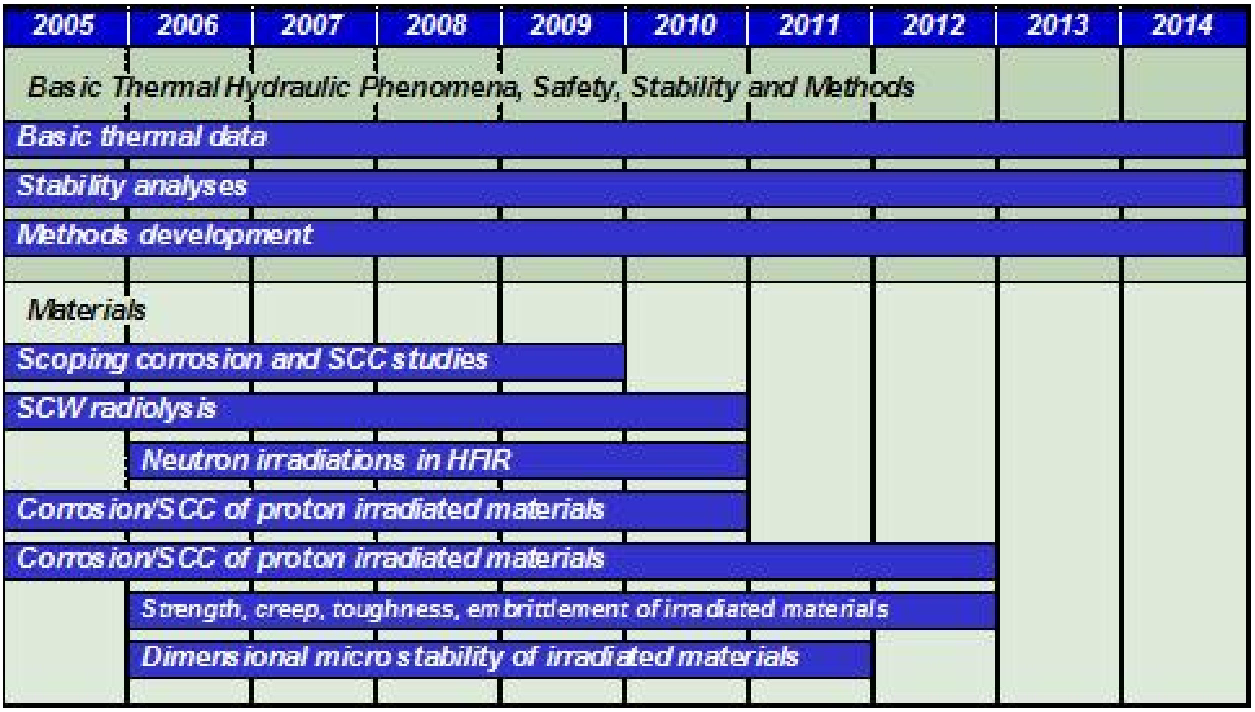
Table 1 shows the SCWR required budget.

Table 1. Required SCWR Budget ($K).

Functional Area	FY-05	FY-06	FY-07	FY-08	FY-09	FY-10	FY-11	FY-12	FY-13	FY-14	Total
Systems Design*	460										
Fuels	0										
Energy Conversion	0										
Materials	466										
TOTAL	**926**										

* Budgets for 2009-2012 include funding for U.S. participation in an international integral facility program.

A2.4.2 10-yr Project Schedule



A2.4.3 10-yr Project Milestones

<u>FY 2005</u>

- Sign and establish a GIF projects in three areas: System Design; Thermal-Hydraulic Phenomena, Safety, Stability, and Methods; and Materials
- Complete design of the test-section for the Erlangen facility
- Complete and put in operation an irradiated samples test facility for stress corrosion cracking testing.

<u>FY 2006</u>

- Complete first set of comparative analyses
- Complete construction and shipment of the test-section for the Erlangen facility
- Complete first PIRT
- Modify a CFD code for SCWR analyses

<u>FY 2007</u>

- Complete stability experiments

- Complete chocked-flow experiments
- Couple a CFD code with RELAP5 for SCWR analyses

FY 2009 and 2010

- Complete SCW heat transfer experiments in Erlangen
- Complete stability analysis
- Establish and participate in an international integral facility project
- Revise PIRT
- Complete corrosion and SCC testing of primary candidate materials for core support components in supercritical water at simulated in-reactor chemistry
- Perform flow-assisted corrosion (FAC) and corrosion fatigue testing for valve materials in supercritical water at simulated chemistry
- Complete collection and evaluation of solid particle erosion in supercritical steam from fossil experience.
- Complete corrosion and SCC screening tests of unirradiated materials in supercritical water

FY 2011 and beyond

- Complete development and assessment of predictive analytical tools for prototypical SCWR conditions
- Complete irradiation of candidate materials in a supercritical pumped flow loop, post-irradiation mechanical properties testing, microstructural characterization, and corrosion and irradiation-assisted stress corrosion cracking tests in supercritical water
- Complete dimensional and microstability of material.

Addendum 1: The U.S. Reference Design

SCWR PRESSURE VESSEL

The reference power, efficiency, pressure, and coolant flow rate and temperatures are listed in Table A2.Addm-1.1. Figure A2.Addm-1.1 is a sketch of the RPV and internals showing the coolant flow paths.

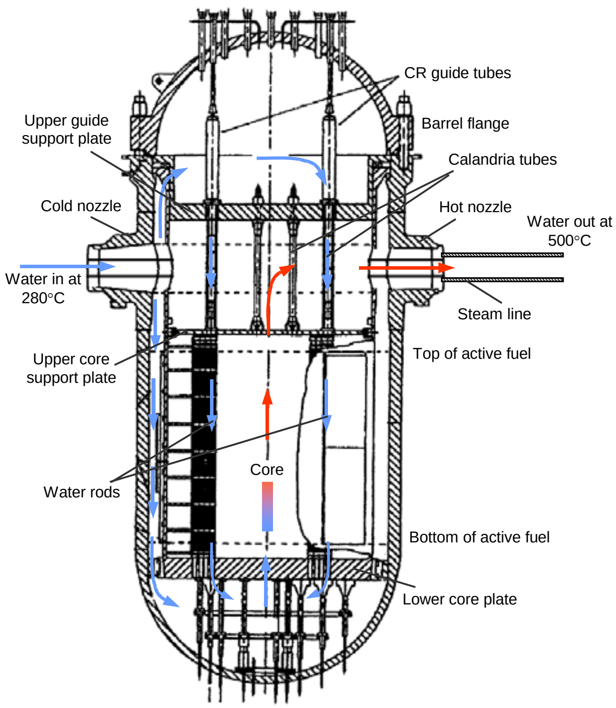


Figure A2.Addm-1.1. The SCWR reactor pressure vessel.

Table A2.Addm-1.1. U.S. Generation-IV SCWR reference design power and coolant conditions.

Parameter	Value
Thermal power	3575 MWt
Net electric power	1600 MWe
Net thermal efficiency	44.8%
Operating pressure	25 MPa
Reactor inlet temperature	280°C
Reactor outlet temperature	500°C
Reactor flow rate	1843 kg/s
Plant lifetime	60 years

Key dimensions for the current SCWR vessel are listed in Table A2.Addm-1.2. The vessel will be exposed to 280°C inlet coolant on the inside surfaces. The outlet nozzles will be protected with a 2" thermal sleeve, which maintains the nozzles below 350°C. Peak fluence of the RPV is expected to be no more than 5×10^{19} n/cm^2 (E>0.1MeV).

Table A2.Addm-1.2. Reference reactor pressure vessel design for the U.S. Generation-IV SCWR

Parameter	Value
Type	PWR with top CRDs
Height	12.40 m
Material	SA-508
Operating/design press.	25.0/27.5 MPa
Operating/design temperature	280/371°C
# of cold/hot nozzles	2/2
Inside diameter of shell	5.322 m
Thickness of shell	0.46 m
Inside diameter of head	5.352 m
Thickness of head	0.305 m
Vessel weight	780 t
Peak fluence (>1 MeV)	$<5\times10^{19}$ n/cm^2

SCWR CORE AND FUEL ASSEMBLY DESIGN

The reference SCWR core design is shown in Figure A2.Addm-1.2. The core will have 145 assemblies with an equivalent diameter of about 3.9 meters. The core barrel will have inside and outside diameters of about 4.3 and 4.4 meters, respectively.

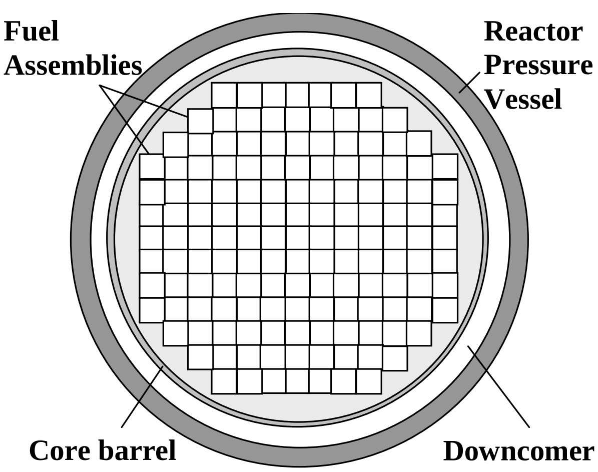


Figure A2.Addm-1.2. Sketch of the reference SCWR core.

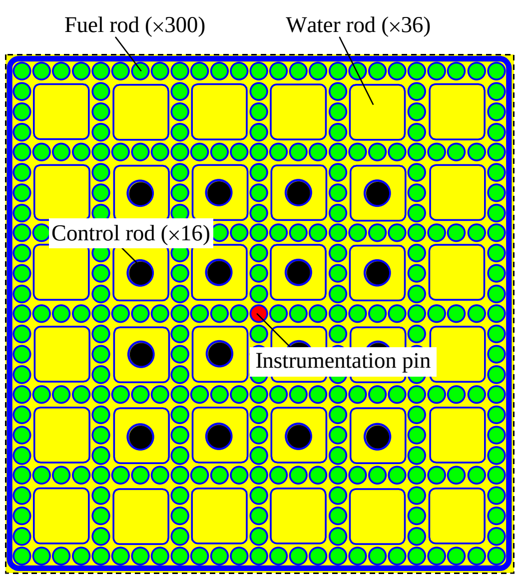


Figure A2.Addm-1.3. The SCWR fuel assembly with water rod boxes.

The reference SCWR fuel assembly design is shown in Figure A2.Addm-1.3 and the relevant dimensions are listed in Table A2.Addm-1.3. It may be necessary to insulate the water rod boxes to retain sufficient moderator density as well as portions of the vessel internals supplying water to the core.

Table A2.Addm-1.3. Reference fuel assembly design for the U.S. Generation-IV SCWR

Parameter	Value
Fuel pin lattice	Square 25x25 array
Number of fuel pins per assembly	300
Number of water rods per assembly	36
Water rod side	33.6 mm
Water rod wall thickness	0.4 mm (plus insulation if needed)
Number of instrumentation rods per assembly	1
Number of control rod fingers per assembly	16
Active control rod materials	B_4C for scram, Ag-In-Cd for control
Number of spacer grids	14 (preliminary estimate)
Assembly wall thickness	3 mm (plus insulation if needed)
Assembly side	286 mm
Inter-assembly gap	2 mm
Assembly pitch	288 mm
Average power density	69.4 kW/L
Average linear power	19.2 kW/m
Peak linear power at steady-state conditions	39 kW/m

The reference fuel pin dimensions are listed in Table A2.Addm-1.4. With the exception of the plenum length and fill pressure, the fuel pin dimensions are typical of 17 by 17 PWR fuel assembly pins. However, the fuel pin pitch is considerably smaller than the pitch used in LWRs. The U-235 enrichment, the Gd_2O_3 loading and fuel burnup are typical of the values used in high burnup LWR fuel.

Table A2.Addm-1.4. Reference fuel pin dimensions for the U.S. Generation-IV SCWR

Parameter	Value
Fuel pin outside diameter	10.2 mm
Fuel pin pitch	11.2 mm
Cladding thickness	0.63 mm
Heated length	4.27 m
Fission gas plenum length	0.6 m
Total fuel pin height	4.66 m
Fill gas pressure at room temperature	6.0 MPa

SCWR PRESSURE VESSEL INTERNALS

The important RPV internals include the lower core support plate, the core former, the core barrel, the upper core support plate, the calandria tubes located immediately above the upper core support plate, the upper guide support plate, the hot nozzle thermal sleeve or insulation, and the control rod guide tubes. The location and approximate shape of most of these components is shown in Figure A2.Addm-1.1.

Some of these components, including the lower core support plate and the control rod guide tubes in the upper head, will be subjected to normal PWR coolant temperature conditions and will be similar to the components typically used in PWRs. However, a number of the RPV internals, including the core barrel (or possibly the core former), the upper guide support plate, the calandria tubes, and the RPV hot nozzle sleeve will be in contact with water at the inlet temperature of 280°C on one side and water at the outlet temperature of 500°C on the other side.

SCWR COOLANT SYSTEM

The SCWR reactor coolant system has two feedwater lines and two steam lines. The main parameters of the SCWR reactor coolant system are listed in Table A2.Addm-1.5.

Table A2.Addm-1.5. SCWR reactor coolant system parameters.

Parameter		Value
Feedwater lines	Number	2
	Operating temperature	280°C
	Operating/design pressure	25/27.5 MPa
	OD/thickness	400 mm / 51 mm
Steam lines	Number	2
	Operating temperature	500°C
	Operating/design pressure	25/27.5 MPa
	OD/thickness	470 mm / 51 mm

SCWR POWER CONVERSION SYSTEM

The reference SCWR system will have a power conversion cycle that is very similar to a supercritical coal-fired plant, with the boiler replaced by the nuclear reactor. The power conversion cycle is based on a large, single-shaft turbine with one high-pressure unit, one intermediate-pressure unit, and three low-pressure units operating at reduced speed (1800 rpm). The steam parameters at the high-pressure/ intermediate-pressure unit inlets are 494°C and 23.4 MPa, well within current capabilities of fossil plants. Similar to traditional light water reactor cycles, a moisture separator-reheater module is located between the high-pressure, intermediate-pressure, and the low-pressure turbines, and reheating is achieved with live nuclear steam. Heat rejection occurs in traditional natural-draft cooling towers. Eight feedwater heaters raise the condensate temperature to the reactor inlet level of 280°C. The main feedwater pumps are turbine-driven and operate at about 190°C.

Addendum 2: SCWR Materials Requirements

REACTOR PRESSURE VESSEL (RPV) MATERIALS

The inner surface of the vessel will be exposed to water at 280ºC. Thus, it would be clad with a weld overlay of Type 308 stainless steel, and the outer surface will be insulated, most likely in a manner similar to existing PWRs. Given the operating temperature of 280ºC and an expected irradiation exposure similar to that of current generation PWR, the primary candidate materials for the RPV shell are those currently used in PWRs, namely SA 508 Grade 3 Class 1 forging (formerly designated SA 508 Class 3) or SA 533 Grade B Class 1 plate. The RPV thickness given in Addendum 1 assumes one of these materials. Of these two materials, which have similar chemical compositions and the same design stress intensities in the American Society of Mechanical Engineers (ASME) Code, the SA 508 Grade 3 Class 1 forging is preferred to eliminate the need for axial welds. It is also desirable to fabricate a forging of sufficient height to keep circumferential welds outside the region adjacent to the reactor core (the so-called beltline region), and preliminary information from the Japan Steel Works indicates that it will probably be possible to do so.

The knowledge gained over the past few decades regarding the radiation embrittlement of current LWR materials must be utilized in the preparation of the material specifications for the RPV materials. For example, minimization of sensitizing elements such as copper and phosphorus is critical for mitigation of embrittlement and undesirable segregation. In addition, the nickel content should be kept relatively low yet high enough to maintain the strength and fracture toughness of the A508 Grade 3 Class 1 steel. In this regard, the thickness of the SCWR vessel shell and nozzle course forgings may present difficulties. Therefore, special attention must be paid to the chemical composition and heat treatment specifications for these two forgings to allow through-thickness hardening to maintain the necessary strength and fracture toughness yet ensure minimization of radiation embrittlement sensitivity.

Similar to the RPV shell, the RPV bolted closure head and welded bottom head will operate at 280ºC, and the materials of construction will be similar. The materials and fabrication of the heads, including the control rod drive mechanism housings, head bolts, etc., will incorporate the latest materials of choice for current LWRs and advanced LWRs. Information regarding RPV supports is not yet available, and the choice of materials will depend upon the specific design.

RPV INTERNALS MATERIALS

Three factors will affect the properties and choice of the structural materials for the fabrication of the SCWR RPV internals. These factors are the effects of irradiation, high-temperature exposure, and interactions with both the sub- and supercritical water environment to which they are exposed. An extensive testing and evaluation program will be required to assess the effects that these factors have on the properties of the potential materials for SCWR construction to enable a preliminary selection of the most promising materials to be made and to then qualify those selected for the service conditions required. Tables A2.Addm-2.1 and A2.Addm-2.2 identify the performance requirements (i.e., the anticipated irradiation conditions and mechanical loads for normal operating conditions as well as the temperature excursions expected for abnormal conditions) and candidate materials for the fuel assembly components and other vessel internals, respectively. The first category includes the fuel cladding, fuel rod spacers (spacer grid or wire wrap), water rod boxes, fuel assembly ducts, and control rod guide thimbles. The second category includes control rod guide tubes, the upper guide support plate, calandria tubes, upper core support plate, lower core plate, core former, core barrel, and threaded structural fasteners. Also listed are materials typical of those in use for similar components in currently operating PWRs and BWRs.

Once a limited set of candidate alloys will be down selected for the cladding, a series of tests will be done to evaluate the safety limits of the fuel pin. Pressure burst and ballooning tests will simulate the fuel pin behavior during depressurization following a large LOCA; rapid heat-up tests will be needed to simulate reactivity initiated accidents, etc.

PUMP, PIPING, AND VALVE MATERIALS

The issues and concerns regarding the pumps, valves, and piping for the SCWR can be divided into those associated with the feedwater lines and the steam lines.

Issues for components of the feedwater system will be similar to those being considered in the more conventional advanced LWR technologies, where ASME Section III is the applicable construction code. Experience has shown that FAC is the dominant degradation mechanism of LWR piping system. In addition, fatigue and stress corrosion cracking are concerns. Carbon steels piping materials in operating LWRs, such as seamless pipe SA-106 Grade C; clad carbon steels; and seamless stainless steels pipes such as SA-312 TP304H, TP304L, TP316L are the primary candidate materials for the feedwater lines. Of these many materials, grades that have been included in the LWR environmental strain-fatigue and fatigue crack growth studies would be preferred. Although seam welded piping has been installed in LWRs, it should be avoided unless the piping has been subsequently reworked and renormalized. Wrought products should be preferred over cast products.

The SCWR feedwater pumps will be low flow/high head pumps located on the feedwater lines outside the containment and are expected to operate at approximately 190°C. These pumps will resemble in many ways state-of-the-art pumps developed for supercritical fossil power. The materials candidates for pump casing are a forged low-alloy steel, such as SA-508 Class 2 or Class 3. An austenitic cladding with controlled delta ferrite content would be required if a low-alloy steel is selected. Alternatively, an austenitic stainless steel such as SA-336 Gr F304 could be considered. The materials candidates for pump internals are a high-strength casting such as SA-487 CA-6NM-A (normalized and tempered 13Cr-4Ni steel).

The steam line piping is the greater concern. The issues related to the steam line system are more akin to those addressed in the design, construction, and operation of supercritical fossil power plants. Creep and time-dependent material degradation are active in fossil-plant steam-line systems at temperatures above 370˚C for ferritic steels and above 425 ˚C for austenitic alloys. The philosophy behind the ASME Power Piping Code (B31.1), which covers fossil plant piping, is significantly different from the philosophy of ASME Section III.

The outlet temperature of 500°C is less than the temperature at which many supercritical fossil power plants operate, but the pressure (25 MPa) is comparable. Whereas ASME Section III has incorporated a wide selection of ferritic piping steels for service to 370 °C and austenitic alloys for service to 425 °C, the high-temperature extension Subsection NH is limited to Grade 22 Class 1, Grade 91, and three austenitic alloys (304H stainless steel, 316H stainless steel, and Alloy 800H). The steam line temperature is sufficiently low to enable the use of one of these materials, providing that FAC is not a problem. Alternate materials would include 316FR stainless steel. This steel qualifies as an “L” grade, yet has properties equivalent to or superior to Type 316H stainless steel. The database is sufficient to meet the needs for inclusion into Subsection NH.

The steam line piping system between the isolation valve and the turbine could be designed to meet the requirements of B31.1, which would allow a greater choice of materials. For example, it would allow the use of alloy P92 (9Cr-2W), which is used in fossil-fired supercritical plants. However, supplementary requirements to address fatigue and other damage accumulation mechanisms would be needed.

EXTERNAL COMPONENTS MATERIALS

Turbine problems have been one of the three leading causes of outages of fossil-fired and nuclear power plants. The main materials causes of these outages have involved thermal fatigue cracking of rotors and discs, condensate-related corrosion or stress corrosion cracking of the last stages of the turbine, and solid particle erosion of the first stage guide vanes.

Attempts to correlate the susceptibility to SCC to alloy microstructural differences (segregation/temper embrittlement) in rotors and discs resulting from the initial metallurgical processing or to the operating history of the turbine have not provided much guidance. SCC occurs only in wet steam at crevices or locations where access to the steam is limited, and it depends on the contaminants present in the steam. Steam in fossil-fired units invariably picks up impurities from sources such as condenser/pump leaks, demineralizer/condensate polisher leaks, de-mineralizer breakdown, and from the feedwater and the water treatment chemicals used. Such impurities will deposit from the steam whenever their solubility is exceeded due to changes in steam temperature and pressure. The contaminants most implicated in SCC are usually chlorides, sulfates, hydroxides, and phosphates of sodium and iron.

Since SCWRs are intended to operate essentially continuously, near maximum load, and at temperatures significantly higher than BWRs, it is expected that their potential for solid particle erosion will be similar to that for the present fleet of fossil-fired supercritical steam power plants. The potential for solid particle erosion damage depends on the physical dimensions of the oxide flakes and the frequency of exfoliation events that varies significantly among the alloy types that are used for the upstream piping. Exfoliation is triggered when the stresses in the growing oxide scales exceed some critical value. These stresses result from the thickness of the scale (accommodation between the volume of oxide formed and the volume of alloy consumed) as well as from the mismatch in the coefficients of thermal expansion of the scale and the underlying alloy during cooling from operating temperature. Relationships have been developed for time, oxide-scale thickness, and tendency for scale exfoliation for some of the candidate alloys used in fossil plants, and these can provide guidance on the time at temperature at which exfoliation problems might be expected.

The materials considerations for the SCWR should be based primarily on fossil plant practice, with two caveats:

1. The maximum alloy temperature required in the SCWR is not higher than the maximum alloy temperature allowed in fossil service.

2. The threat of SCC from oxidizing or other species resulting from radiolysis of the water is not greater than that from the water conditions prevailing in the supercritical fluid in fossil plants.

Table A2.Addm-2.1. Operating conditions and candidate materials for the in-core reactor components of the SCWR. All components listed are part of replaceable fuel assembly.

Component	Normal Conditions			Abnormal Conditions	Current LWR Mtls		Candidate SCWR Materials	Notes
	Temperature [1]	Peak Dose [2]	Loads [3]	Temperature [4]	PWR	BWR		
Fuel cladding	280-620 ºC	15 dpa	Pressure drop across cladding, grid-cladding and fuel-cladding interactions σ up to 100 MPa	Up to 840°C for <30 sec	Zircaloy 4	Zircaloy 2	Fe-Ms, Low-swell S.S.	
Spacer grids/wire wrap	280-620 ºC	15 dpa	Hold the fuel pins together	Up to 840°C for <30 sec	Zircaloy 4, Inconel 718	Zircaloy 4, Inconel X750, 304 S.S.	Fe-Ms, Low-swell S.S.	
Water rod boxes	280-300 ºC inner 280-500 ºC outer	15 dpa	ΔP<0.1 MPa	Up to 700°C for <30 sec	N/A	Zircaloy 2	Fe-Ms, Low-swell S.S.	May need to insulate.
Fuel Assembly duct	280-500 ºC inner 280-300 ºC outer	15 dpa	ΔP<0.1 MPa	Up to 700°C for <30 sec	N/A	Zircaloy 4	Fe-Ms, Low-swell S.S.	May need to insulate.
Control Rod Guide Thimble	280-300 ºC	15 dpa	Low hydraulic and thermal stresses	280 - 300°C	Zircaloy 4	N/A	Zircaloy 4, Zr-Nb alloy	Zr alloy selected for superior neutron economy.
1. Peak temperatures in PWRs are 320-370°C 2. Design estimates for typical high burnup LWR fuel 3. In addition, all reactor internals will be subject to seismic and pipe break loads. 4 Condition II events only (LOCAs, LOFAs, ATWSs are excluded)			Fe-Ms (Ferritic-Martensitic) steels, e.g., T91 (9Cr-1Mo-V), A-21 (9Cr-TiC mod), NF616 (9Cr), HCM12A (12Cr), 9Cr-2WVTa, MA-957. Existing low-swell stainless steels, e.g., D-9 (14.5Cr-14.5Ni, 2Mo, Ti stab), PNC ~D-9 mod w/P).					

Table A2.Addm-2.2. Operating conditions and candidate materials for the core structural support reactor components of the SCWR.

Component	**Normal Conditions**			**Abnormal Conditions**	**Current LWR Mtls**		**Candidate SCWR Materials**	**Notes**
	Temperature [1]	Peak Dose [2]	Loads [3]	Temperature	PWR	BWR		
Upper Guide Support (UGS) plate	280 ºC upper 500 ºC lower	0.021 dpa	Significant hydraulic and thermal loads	Lower side at up to 700°C for <30 sec	304L S.S	304L S.S.	Advanced S.S., Fe-Ms	Must insulate between the region above the core (500°C) and the upper plenum (280°C) to limit the thermal loads in the UGS.
Calandria Tubes	280 ºC inner 500 ºC outer (w/o insulation)	0.021 dpa	Significant hydraulic and thermal loads	280°C inner 700°C outer	N/A	N/A	Advanced S.S., Fe-Ms	Must insulate to limit the heat transfer from the coolant to the moderator and control the thermal loads in the calandria tubes.
Upper Core Support (UCS) plate	500 ºC	0.021 dpa	Significant hydraulic. Moderate thermal.	Up to 700 ºC for <30 sec	304 S.S.	304, 304L, 316 S.S.	Advanced S.S., Fe-Ms	The water rod box penetrations may cause some locally high thermal stresses.
CR guide tubes	280 ºC	0.00001 dpa	Low hydraulic. Low thermal.	N/A	304 S.S.	304 S.S.	Advanced S.S., Fe-Ms, 304L, 316L	May want to use the same material as for the UGS, UCS, and calandria tubes
Lower core plate	280-300 ºC	0.39 dpa	Significant hydraulic. Low thermal. Supports core.	N/A	304L S.S	304L S.S.	Advanced S.S, Fe-Ms, 304L, 316L	May want to use the same material as for the UGS, UCS, and calandria tubes
Core Former	~280-600 ºC	67.1 dpa	Significant hydraulic. High thermal.	700ºC	304 S.S.	N/A	Fe-Ms, Low-Swell S.S.	Must insulate either the core former or core barrel to control the thermal loads in the barrel.
Core barrel or shroud	280ºC core region, 500 ºC above core	3.9 dpa	Significant hydraulic. High thermal.	N/A	304L S.S	304L S.S.	Fe-Ms, Low-Swell S.S.	Must insulate the core barrel above the core region and insulate either the core barrel or core former in the core region.
Threaded fasteners	280-500 ºC	< 4 dpa [4]			316 S.S./CW	304, 600, 316, 316L	Advanced S.S., IN-718, 625, 690	The current design is an all welded core former and barrel.

1. Peak temperatures in PWRs are 320-370°C
2. Design estimates for 60y
3. All reactor internals will be subject to seismic and pipe break loads
4. ~ 50 dpa for baffle bolts and formers in PWRs

Fe-Ms (Ferritic-Martensitic) steels, e.g., T91 (9Cr-1Mo-V), A-21 (9Cr-TiC mod), NF616 (9Cr), HCM12A (12Cr), 9Cr-2WVTa, MA-957.

Existing low-swell stainless steels, e.g., D-9 (14.5Cr-14.5Ni, 2Mo, Ti stab), PNC ~D-9 mod w/P). Advanced stainless steels, e.g., HT-UPS (~PNC), AL-6XN (20Cr-24Ni-6Mo-0.2Cu-0.2N), etc.