Appendix 2.0

Supercritical-Water-Cooled Reactor

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ACRONYMS

ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BOP	balance-of-plant
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium
CFD	computational fluid dynamics
DOE	Department of Energy
CO_2	carbon dioxide
dpa	displacements per atom
FAC	flow-assisted corrosion
FFP	fossil-fuel plants
FY	fiscal year
GIF	Generation IV International Forum
I-NERI	International Nuclear Energy Research Initiative
INL	Idaho National Laboratory
LOCA	loss of coolant accident
LWR	Light Water Reactor
MPa	megapascals
MWe	megawatt electric
MWt	megawatt thermal
NERI	Nuclear Energy Research Initiative
PHWR	Pressurized Heavy Water Reactor
PWR	Pressurized Water Reactor
R&D	research and development
RPV	Reactor Pressure Vessel
SCC	stress corrosion cracking
SCW	supercritical water
SCWO	supercritical water oxidation
SCWR	Supercritical-Water-Cooled Reactor

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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% versus 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. A conceptual SCWR system is shown in Figure A2.1. 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, pressurizers, steam generators, and steam separators and dryers in current LWRs is eliminated.

The main mission of the SCWR is generation of low-cost electricity (note that the SCWR begins with a thermal neutron spectrum and once-through fuel cycle, but may ultimately be able to achieve a fast-spectrum with recycle). 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 system 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 so it will not be repeated here.

In fiscal year (FY) 2006, the SCWR program was redirected. The current plan focuses on materials research needs to be conducted to establish the optimal operational parameter range for SCWR from a materials point of view and ensure selection of structural and cladding materials that will maintain reliable operation of a SCWR power plant for its design life.

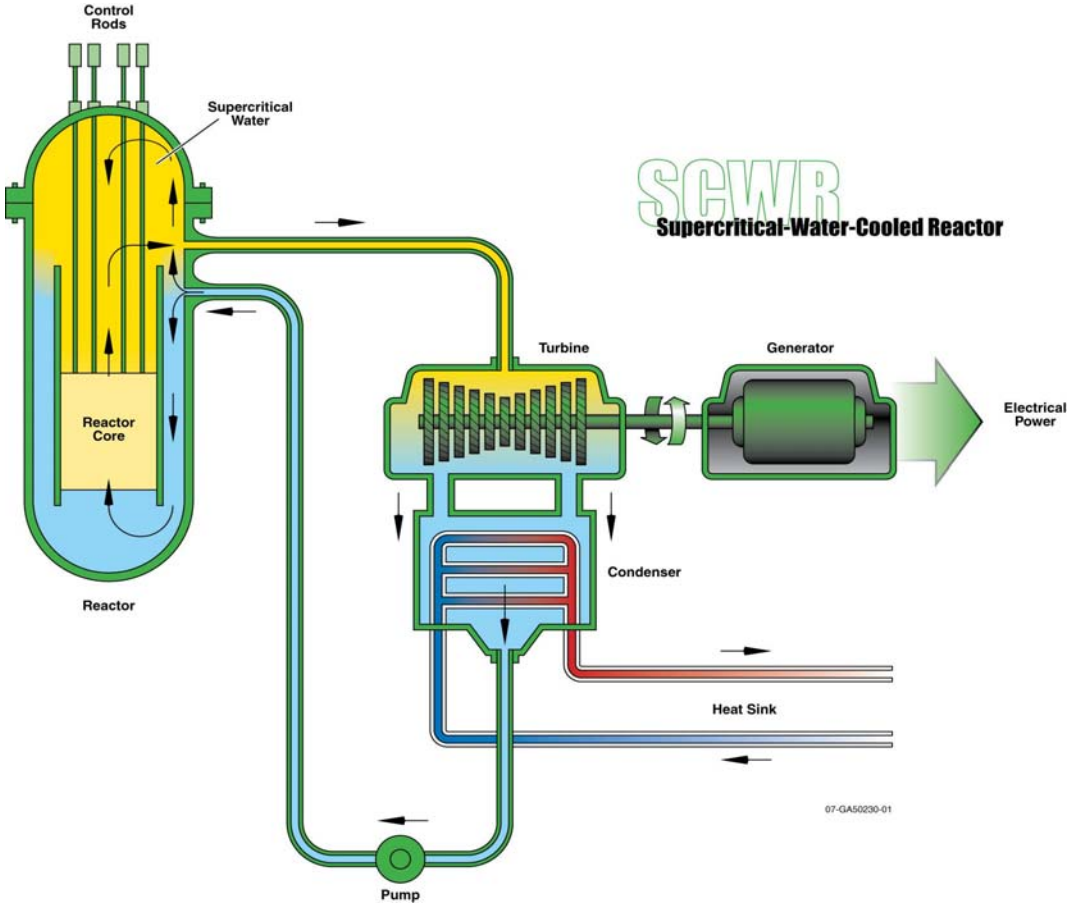


Figure A2.1. Conceptual SCWR system.

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 their approach to the reactor design: one utilizes a reactor pressure vessel and the other utilizes pressure tubes. The main difference between these pressure vessel concepts lies in the core layout and the moderators. 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, Republic of Korea, and Europe is focused on the pressure vessel concept. Canada selected a pressure tube design for its SCWR as the logical evolution of Canada deuterium uranium (CANDU)-type reactors. The U.S. Generation IV SCWR Program operates under the following system characteristics, 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 system characteristics 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 before flowing 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 Addendum A2-1 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. If fueled loop testing is required, the schedule may be extended. The SCWR Steering Committee sponsored an expert's workshop in March 2006 at Nuclear Energy Agency Headquarters in Paris, France, to address this issue. Topics discussed included (1) whether testing is needed and (2) if it is needed, what facilities are available to perform the testing? The conclusion of the workshop was that in-pile fuel testing would be necessary for any combination of cladding material and fuel composition if a SCWR was to ever be licensed.

A2.2 RESEARCH AND DEVELOPMENT STRATEGY

A2.2.1 Objectives

The original objective of this ten-year program plan was to identify activities needed to assess the technical feasibility of the SCWR concept that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance. This ten-year period roughly corresponds to the duration of the viability assessment phase as envisaged in *A Technology Roadmap for Generation-IV Nuclear Energy Systems* (DOE 2002). This document 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 are 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 flow sheets established through testing at appropriate scale
3. Cost analysis based on preconceptual design
4. Simplified probability risk assessment 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 define the strategy and the framework for the scope of the overall R&D that needs to be conducted. Item 5 is the endpoint that this revision of the research plan addresses. The other endpoints depend on a specific design, which will be selected by the GIF partners.

The initial work on SCWR conducted in the U.S. and elsewhere resulted in the 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 (BOP) design, reactor vessel design, startup procedure development, and stability analysis. The work pointed to the need for further development of specific thermal-hydraulic databases for development and validation of analytical tools. It also indicated that core design optimization is needed to ensure reliable operation of the reactor under normal operating conditions. Because of the initial SCWR program redirection in FY 2005, which focused on R&D activities rather than plant design, and a subsequent redirection in 2006, to further focus R&D efforts on materials research, the primary R&D focus will be on materials research. SCWR viability will be addressed using the U.S. baseline concept design generated in 2003-2004.

A2.2.2 Scope

The original SCWR R&D program scope is organized and based on the issues identified in the Roadmap (DOE 2002). Its three main elements 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 from 2001 to 2004 through NERI and Generation IV Program activities. A preconceptual 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 downflow 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 BOP was conceptually designed and analyzed, and potential startup procedures were developed.

However, because plant design activities were redirected in 2005 and 2006, research activities will focus on addressing current, basic knowledge gaps in materials and chemistry issues. The key to demonstrating the viability of the SCWR design is to identify and develop materials that can ensure 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.

A2.2.3 Viability Issues

The Generation IV Technology Roadmap (DOE 2002) identified availability of reliable materials as a key viability issue.

The identification of appropriate materials for in-core and out-core components and understanding of supercritical water (SCW) chemistry are two 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 a SCWR. Although there is considerable experience with fast reactors and supercritical-water-cooled fossil-fueled plants (FFPs), there are 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 SCW. The marked change in the density of 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 similar to LWR's in many aspects; 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 the LWR response in accident conditions. Therefore, demonstration of adequate levels of SCWR safety and availability of reliable analytical tools are critical viability issues.

Additionally, due to the large 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 ensure 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 Generation IV International Forum Research and Development Projects

Due to changing priorities in the Generation IV Program, the U.S. will no longer be participating in GIF related SCWR activities and will not be a signatory to the SCWR System Arrangement. However, technical information exchanges will still take place with SCWR GIF countries as appropriate. Several material compatibility and the radiolysis behavior issues need to be more fully addressed.

A2.2.4.2 University Collaborations

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

The key participating universities and their areas of focus are:

- Massachusetts Institute of Technology—stability analyses (completed)
- University of Michigan—materials research
- University of Wisconsin—materials research
- University of Notre Dame—materials research and chemistry
- Rensselaer Polytechnic Institute—thermal-hydraulic database 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. Burns and Roe (2003-2004) provided support in BOP design, startup, and development of control procedures. Both industrial partners provided substantial contribution to the initial conceptual system design and safety aspects of the plant. These projects have been completed and there are currently no active industry partner interactions.

A2.2.4.4 International Nuclear Energy Research Initiative/Nuclear Energy Research Initiative

Two NERI projects, launched in 2001, included significant design activities. One is led by the Idaho National Laboratory (INL) and involves Westinghouse Electric Company, and the other is led by the University of Wisconsin at Madison and involves Argonne National Laboratory. Both projects have produced significant modifications to the Japanese and European baseline designs. Two I-NERI projects were also initiated in 2005. There is an I-NERI project with Korea which focuses on a thermal-hydraulic database, and an I-NERI with project Canada, which focuses both on a thermal-hydraulic database and materials. In 2005, a NERI project was initiated with Rensselaer Polytechnic Institute to utilize their NPHASE computational fluid dynamics code for analysis of SCW experiments.

A2.3 HIGHLIGHTS OF RESEARCH AND DEVELOPMENT

A2.3.1 System Design

This R&D element provides the pre-conceptual SCWR design needed for the viability assessment and guidance for materials and chemistry, 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. Work performed by the GIF partners from 2007 to 2015 will focus on identification of the most promising design. Since 2005, the U.S. has not actively participated in this activity.

A2.3.1.1 System and Comparative Analyses

The objective of this GIF activity is to converge on a design that can be developed jointly and demonstrated cooperatively by the GIF countries. It will include operational analyses, safety analyses, and economic assessment. The U.S. has not actively participated in this activity since 2004.

A2.3.2 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. The GIF partners will collaborate to conduct the necessary experiments, develop databases, and address and improve analytical models and codes where necessary. Codes will be validated against available and planned experimental data and benchmarked against other codes developed by the GIF partners or elsewhere.

A2.3.2.1 Basic Thermal Data for the Supercritical-Water-Cooled Reactor

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 SCW coolant with great accuracy. However, while considerable information exists on heat transfer to SCW in round tubes for fossil boilers, little is known about the effect of the geometry of rod bundles and fluid conditions typical of the SCWR core. This project was to address the critical issue of measuring heat transfer to SCW at prototypical SCWR conditions and developing the tools to predict the SCWR thermal behavior. However, in 2006, this effort was brought to a conclusion to allow the project to

concentrate on SCWR materials issues. A design of a test section was completed to allow for bundle heat transfer tests at typical SCWR conditions. This design information was made available to SCWR GIF members. The test section is shown in Figure A2.2

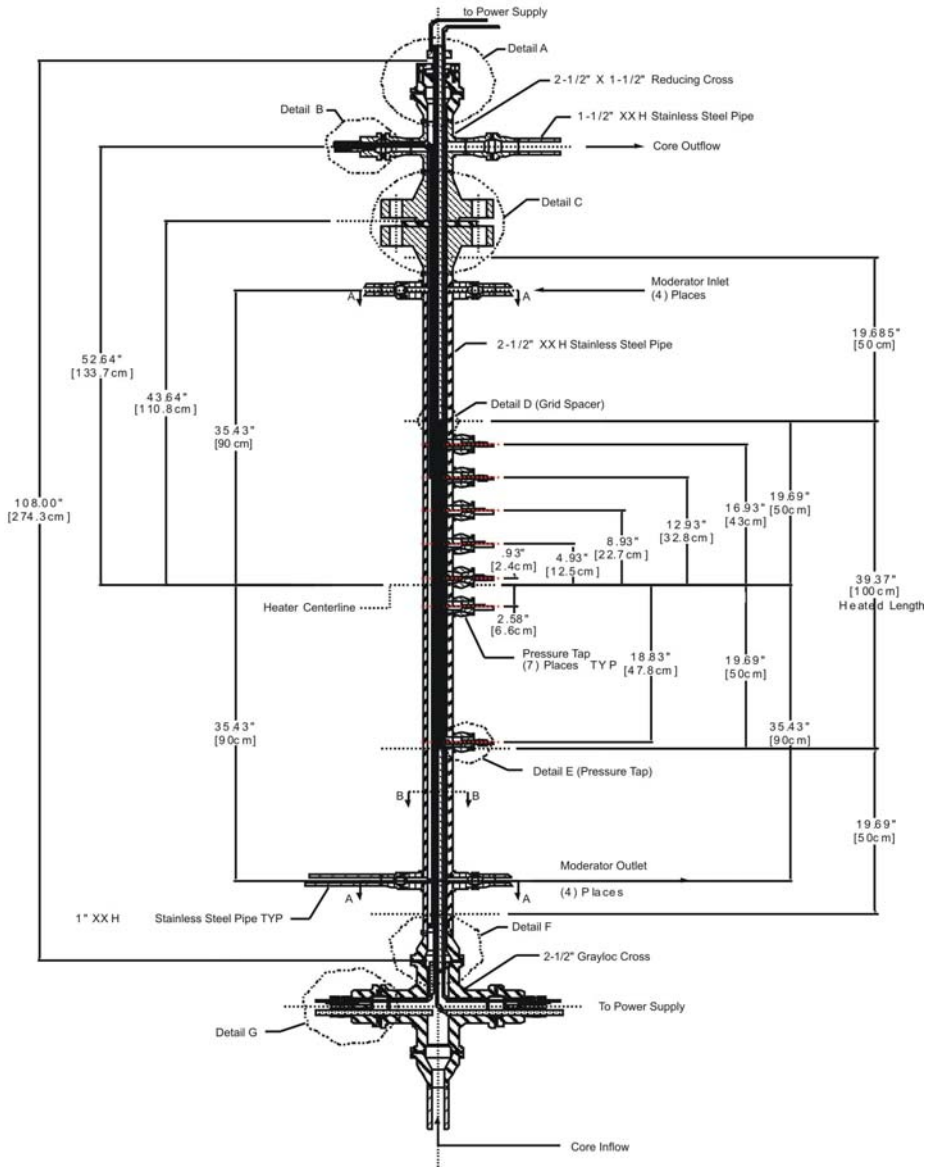


Figure A2.2. Bundle heat transfer test section.

A2.3.2.2 Supercritical-Water-Cooled Reactor 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 due to these instabilities do not occur during normal operation or, if they do, that they can be detected and suppressed in a safe manner. Oscillations under accident conditions must also be considered, e.g., under anticipated transient without scram conditions. The objective of this task is to achieve 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.

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.

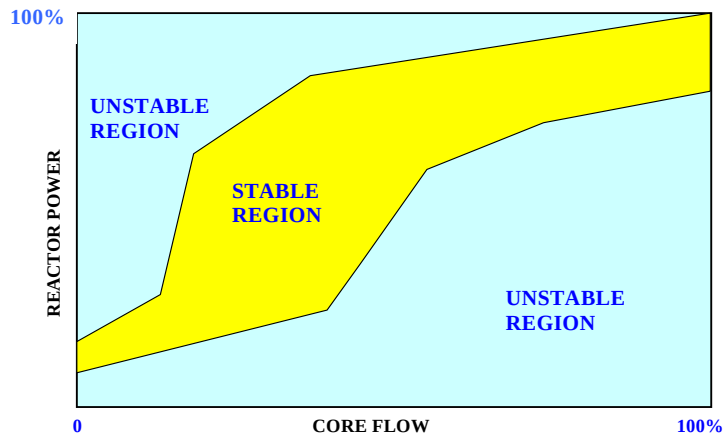


Figure A2.3. Conceptual envelope for stable power-flow operation of SCWR.

A2.3.3 Fuels

The SCWR system is based on standard LWR fuel. The SCWR Steering Committee sponsored an expert's workshop in March 2006 at Nuclear Energy Agency Headquarters in Paris, France, to address this issue. Topics discussed included (1) whether testing is needed and (2) if it is needed, what facilities are available to perform the testing? The conclusion of the workshop was that in-pile fuel testing would be necessary for any combination of cladding material and fuel composition if a SCWR was to ever be licensed.

A2.3.4 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 BOP and turbine issues can be resolved and are not a problem.

A2.3.5 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 SCWR materials requirements in greater detail.

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, may 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 has currently received enough study to unequivocally ensure its viability in a SCWR. An extensive review of potential materials is not presented here but, based on experiences from LWRs, pressurized heavy water reactors (PHWRs), fast reactors, and SWC 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 to 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 a SCWR environment.

There are little data on the general corrosion behavior of any of the candidate materials in SCW or at temperatures between current water reactors and the pseudo-critical temperature. Below the pseudo-critical temperature, the density of water is similar to that of a conventional liquid, so the chemistry is expected 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 SCW 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, intergranular attack, intergranular 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. SCC has been observed in supercritical water facilities 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 a SCWR will be an enhanced understanding of the chemistry of SCW. 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 than at the outlet. These complications are further exacerbated by in-core radiolysis.

Understanding the corrosion potential in a SCWR core and effectively controlling corrosion rates 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-core and out-core. Thermal and irradiation creep will be higher at SCW temperatures and pressures, although the enhancement in creep rate between that observed in conventional water-cooled reactors and conditions equivalent to a 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 such that that its integrity may be questionable during (and/or following) an outage. In addition, fatigue and thermal aging effects on material properties 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, He-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 are 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 are 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 PROJECT COST AND SCHEDULE

A2.4.1 Fiscal Year 2007 Project Budget

Table A2.1. FY 2007 budget profile for SCWR activities ($K).

Task	FY-07[a]
Materials	500
Total	500

a. FY 2007 budget includes FY 2006 carryover funds.

A2.4.2 Ten-Year Project Schedule

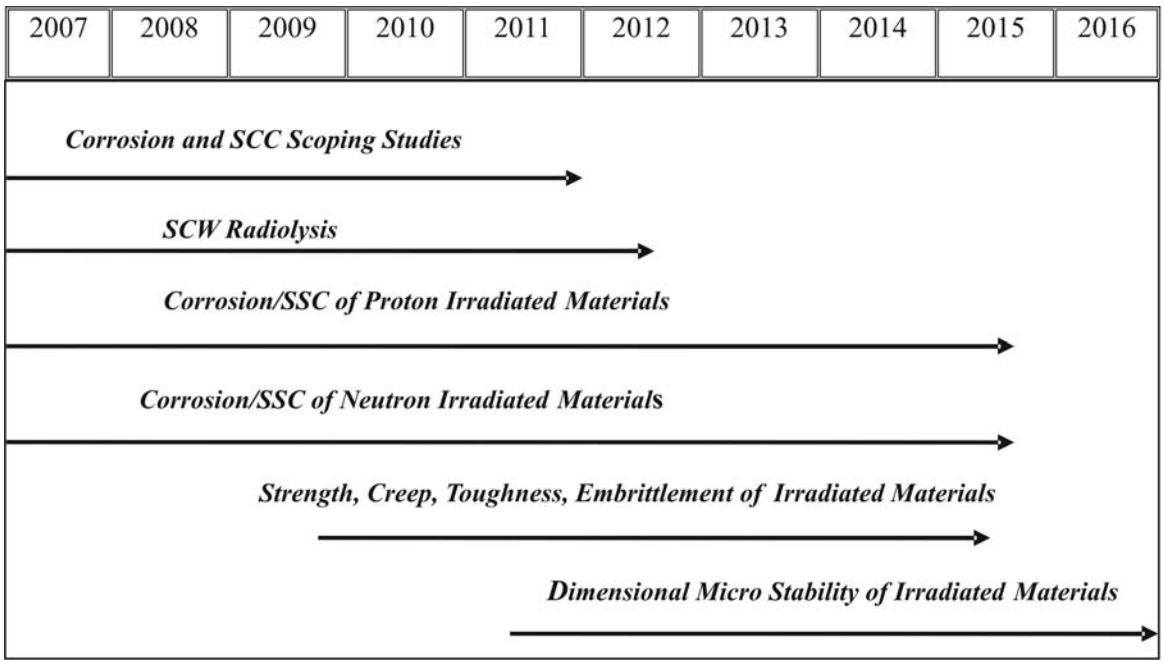


Figure A2.4. SCWR development timeline.

A2.4.3 Ten-Year Project Milestones

FY 2007

- Complete first neutron irradiated materials test
- Complete study on using DOE university facilities as user facilities.

FY 2008

- Complete baseline crack growth studies in SCW.

FY 2009 – 2010

- Complete corrosion and SCC screening tests of unirradiated materials in SCW.

FY 2011 and beyond

- Complete neutron irradiated materials testing and analysis
- Complete dimensional and microstability testing of material.

A2.5 REFERENCES

DOE, 2002, *A Technology Roadmap for Generation-IV Nuclear Energy Systems*, GIF-002-00, U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, December 2002.

ADDENDUM A2-1: THE U.S. REFERENCE DESIGN

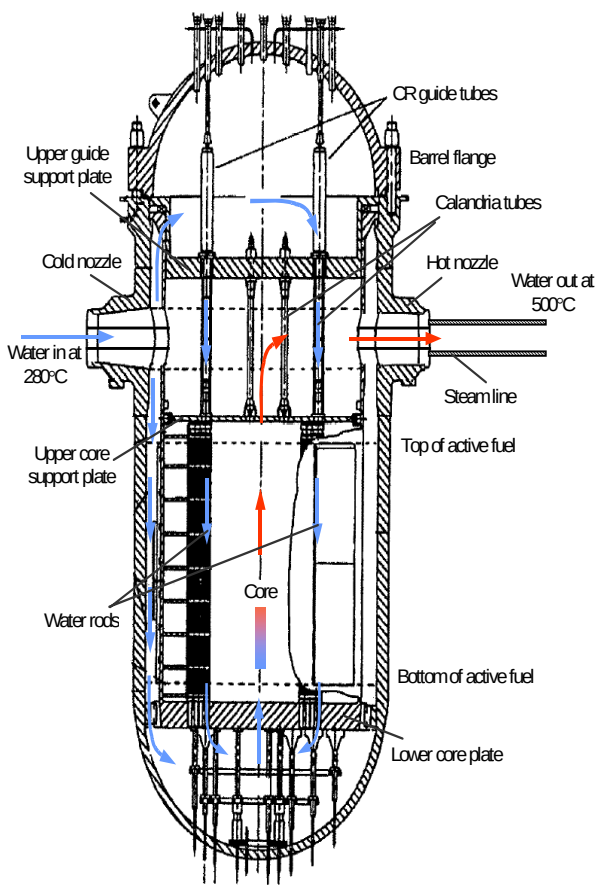
Supercritical-Water-Cooled Reactor Pressure Vessel

Addm A2-1: Table 1 lists the reference power, efficiency, pressure, and coolant flow rate and temperatures. Addm A2-1: Figure 1 is a sketch of the reactor pressure vessel (RPV) and internals showing the coolant flow paths.

Key dimensions for the current SCWR vessel are listed in Addm A2-1: Table 2. The vessel will be exposed to 280°C inlet coolant on the inside surfaces. The outlet nozzles will be protected with a 2-in. 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).

Addm A2-1: Table 1. U.S. Generation IV SCWR reference design power and coolant conditions.

Parameter	Value
Thermal power	3,575 MWt
Net electric power	1,600 MWe
Net thermal efficiency	44.8%
Operating pressure	25 MPa
Reactor inlet temperature	280°C
Reactor outlet temperature	500°C
Reactor flow rate	1,843 kg/s
Plant lifetime	60 years



Addm A2-1: Figure 1. The SCWR reactor pressure vessel.

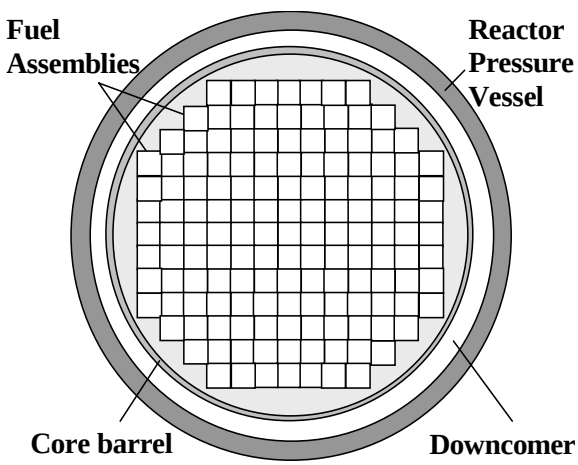
Addm A2-1: Table 2. Reference reactor pressure vessel design for the U.S. Generation IV SCWR.

Parameter	Value
Type	Pressurized water reactor (PWR) with top control rod drives (CRDs)
Height	12.40 m
Material	SA-508
Operating/design pressure	25.0/27.5 MPa
Operating/design temperature	280/371°C
Number 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)	$<5 \times 10^{19}$ n/cm^2

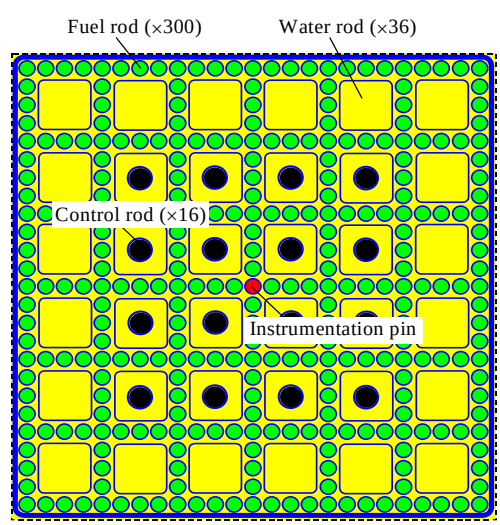
Supercritical-Water-Cooled Reactor Core and Fuel Assembly Design

The reference SCWR core design is shown in Addm A2-1: Figure 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.

The reference SCWR fuel assembly design is shown in Addm A2-1: Figure 3 and the relevant dimensions are listed in Addm A2-1: Table 3. It may be necessary to insulate the water rod boxes as well as portions of the vessel internals supplying water to the core to retain sufficient moderator density.



Addm A2-1: Figure 2. Sketch of the reference SCWR core.



Addm A2-1: Figure 3. The SCWR fuel assembly with water rod boxes.

Addm A2-1: Table 3. Reference fuel assembly design for the U.S. Generation IV SCWR.

Parameter	Value
Fuel pin lattice	Square 25 × 25 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
Interassembly 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 Addm A2-1: Table 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.

Addm A2-1: Table 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

Supercritical-Water-Cooled Reactor 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 Addm A2-1: Figure 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.

Supercritical-Water-Cooled Reactor 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 Addm A2-1: Table 5.

Addm A2-1: Table 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

Supercritical-Water-Cooled Reactor 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 (1,800 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 LWR cycles, a moisture separator-reheater module is located between the high-pressure, intermediate-pressure, and 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 A2-2: SUPERCRITICAL-WATER-COOLED REACTOR MATERIALS REQUIREMENTS

Reactor Pressure Vessel Materials

The inner surface of the vessel will be exposed to water at 280ºC. Thus, it will 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 PWRs, 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) Boiler and Pressure Vessel (B&PV) Code (Section 2, Part D, Subpart 1, Stress Tables), 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 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.

Reactor Pressure Vessel 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 SCW 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 6 and 7 in this Addendum 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 is 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 loss-of-coolant accident (LOCA); rapid heat-up tests will be needed to simulate reactivity-initiated accidents, etc.

Addm A2-2: Table 1. Operating conditions and candidate materials for the in-core 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 [4]	PWR	BWR		
Fuel cladding	280-620ºC	15 displacements per atom (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,[5] Low-swell S.S. [6]	
Spacer grids/ wire wrap	280-620ºC	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,[5] Low-swell S.S. [6]	
Water rod boxes	280-300ºC inner 280-500ºC outer	15 dpa	ΔP<0.1 MPa	Up to 700°C for <30 sec	N/A	Zircaloy 2	Fe-Ms,[5] Low-swell S.S.[6]	May need to insulate.
Fuel Assembly duct	280-500ºC inner 280-300ºC outer	15 dpa	ΔP<0.1 MPa	Up to 700°C for <30 sec	N/A	Zircaloy 4	Fe-Ms,[5] Low-swell S.S.[6]	May need to insulate.
Control Rod Guide Thimble	280-300ºC	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).
5. Fe-Ms (Ferritic-Martensitic) steels, e.g., T91 (9Cr-1Mo-V), A-21 (9Cr-TiC mod), NF616 (9Cr), HCM12A (12Cr), 9Cr-2WVTa, MA-957.
6. Existing low-swell stainless steels, e.g., D-9 (14.5Cr-14.5Ni, 2Mo, Ti stab), PNC ~D-9 mod w/P).
*All components listed are part of replaceable fuel assembly.

Addm A2-2: Table 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	280ºC upper 500ºC 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.,[5] Fe-Ms[6]	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	280ºC inner 500ºC 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.,[5] Fe-Ms[6]	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	500ºC	0.021 dpa	Significant hydraulic. Moderate thermal.	Up to 700ºC for <30 sec	304 S.S.	304, 304L, 316 S.S.	Advanced S.S., [5] Fe-Ms[6]	The water rod box penetrations may cause some locally high thermal stresses.
CR guide tubes	280ºC	0.00001 dpa	Low hydraulic. Low thermal.	N/A	304 S.S.	304 S.S.	Advanced S.S., [5] Fe-Ms, [6] 304L, 316L	May want to use the same material as for the UGS, UCS, and calandria tubes
Lower core plate	280-300ºC	0.39 dpa	Significant hydraulic. Low thermal. Supports core.	N/A	304L S.S	304L S.S.	Advanced S.S, [5] Fe-Ms, [6] 304L, 316L	May want to use the same material as for the UGS, UCS, and calandria tubes
Core former	~280-600ºC	67.1 dpa	Significant hydraulic. High thermal.	700ºC	304 S.S.	N/A	Fe-Ms, [6] Low-Swell S.S.[7]	Must insulate either the core former or core barrel to control the thermal loads in the barrel.
Core barrel or shroud	280ºC core region, 500ºC above core	3.9 dpa	Significant hydraulic. High thermal.	N/A	304L S.S	304L S.S.	Fe-Ms, [6] Low-Swell S.S.[7]	Must insulate the core barrel above the core region and insulate either the core barrel or core former in the core region.
Threaded fasteners	280-500ºC	< 4 dpa [4]			316 S.S./CW	304, 600, 316, 316L	Advanced S.S.,[5] 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.
5. Advanced stainless steels, e.g., HT-UPS (~PNC), AL-6XN (20Cr-24Ni-6Mo-0.2Cu-0.2N), etc.
6. Fe-Ms (Ferritic-Martensitic) steels, e.g., T91 (9Cr-1Mo-V), A-21 (9Cr-TiC mod), NF616 (9Cr), HCM12A (12Cr), 9Cr-2WVTa, MA-957.
7. Existing low-swell stainless steels, e.g., D-9 (14.5Cr-14.5Ni, 2Mo, Ti stab), PNC ~D-9 mod w/P).

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 those associated with 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 B&PV Code Section III is the applicable construction code. Experience has shown that flow assisted corrosion (FAC) is the dominant degradation mechanism of LWR piping systems. In addition, fatigue and stress corrosion cracking are concerns. Carbon-steel 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 B&PV Code 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 material 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 of 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, essentially, operate 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 and 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.

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