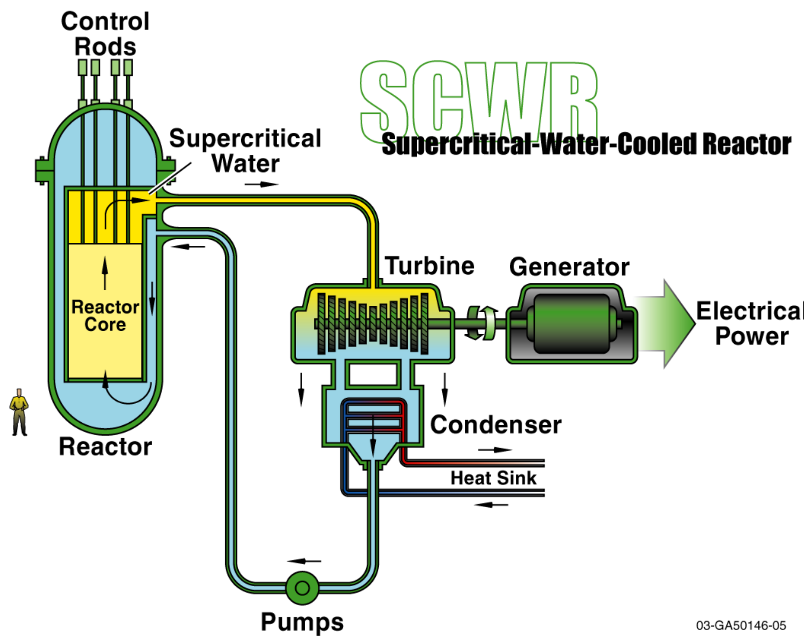
INEEL/EXT-03-01210

Supercritical Water Reactor (SCWR)

Progress Report for the FY-03 Generation-IV R&D Activities for the Development of the SCWR in the U.S.



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September 30, 2003

Executive Summary

Supercritical water cooled reactors (SCWRs) are essentially light water reactors (LWRs) operating at higher pressure and temperature. SCWRs achieve high thermal efficiency (i.e., about 45% vs. about 35% efficiency for advanced LWRs) and are simpler plants as the need for many of the traditional LWR components such as the coolant recirculation pumps, pressurizer, steam generators, and steam separators and dryers is eliminated. SCWRs build upon two proven technologies, the LWR and the supercritical coal-fired boiler. The main mission of the SCWR is production of low-cost electricity. Thus the SCWR is also suited for hydrogen generation with electrolysis, and can support the development of the hydrogen economy in the near term. SCWRs are one of only six reactor technologies currently being studied under the Generation-IV international program.

In FY-03 the Generation-IV SCWR program in the U.S. comprised six tasks involving six organizations, i.e., Idaho National Engineering and Environmental Laboratory (INEEL), Argonne National Laboratory (ANL), Oak Ridge National Laboratory (ORNL), the Westinghouse Electric Company (including the BWR Engineering group in Sweden), Burns & Roe Enterprises Inc. (BREI) and the Massachusetts Institute of Technology (MIT). The total budget for FY-03 was $438k with the cost breakdown reported in Table E.I. The objective of the multi-year SCWR program is to assess the technical viability of the SCWR concept. Thus, as per the guidelines in the Generation-IV Roadmap Report, the focus is on establishing a conceptual design, assessing its safety and stability characteristics, and identifying and testing candidate materials for all reactor components.

Table E.I. Task and cost breakdown for the U.S. Generation-IV SCWR program in FY-03.

Task	Organization	Budget ($k)
Program Management	INEEL	75
Balance of plant design, control and start-up	BREI	50
Stability analysis	ANL	53
Containment and safety systems design	Westinghouse	160
Materials survey	ORNL	50
Corrosion testing of candidate materials	MIT	50
Total		**438**

The team has selected a reference design for the SCWR system that focuses on a large-size, direct-cycle, thermal-spectrum, light-water cooled and moderated, low-enriched uranium fuelled, base-load operation plant for electricity generation at low capital and operating costs. The operating pressure and core inlet/outlet temperatures are 25 MPa and 280/500°C, respectively. The coolant density decreases from about 760 kg/m^3 at the core inlet to about 90 kg/m^3 at the core outlet. Thus, large square water rods with down flow are used to provide adequate moderation in the core. The fuel pin design is similar to that of a pressurized water reactor (PWR), but with higher fill pressure and longer fission gas plenum. Candidate materials for all fuel assembly and vessel internal components have been identified by ORNL and include ferritic-martensitic steels and low-swelling austenitic steels for the components exposed to high neutron doses, and high-strength austenitic steels and nickel-based alloys for low-dose components. However, these materials are un-proven in the potentially aggressive SCWR environment, and their performance will have to be tested. A materials development program has been prepared for this purpose.

Two traditional austenitic steels (304L and 316L) were tested at MIT and the University of Michigan for corrosion and stress-corrosion cracking (SCC) susceptibility in supercritical water. It was found that both alloys are susceptible to SCC (316L less so than 304L) in both deaerated and non-deaerated high-temperature (>400°C) supercritical water. Thus, these alloys cannot be used for high-temperature components in the SCWR. However, they could be used for components operating in the

280-350°C range (e.g., the lower core plate, the control rod guide tubes), given their satisfactory behavior in deaerated water at these temperatures.

The SCWR core average power density is about 70 kW/L, i.e., between the power density of boiling water reactors (BWRs) and PWRs. The core rated thermal power is 3,575 MW resulting in a pressure vessel of 5.3-m inner diameter and 46-cm thickness in the beltline region. The vessel operates at 280°C and traditional LWR low-alloy steels such as SA-508 can be used. Because of its large size the vessel cannot be manufactured in the U.S., but appears to be within existing manufacturing capabilities in Japan. The vessel was sized by Westinghouse per the ASME Code regulations, and its structural performance was verified with detailed 3D finite-element analyses.

The reactor coolant system of the SCWR comprises the feedwater lines and main steam lines up to the outermost set of containment isolation valves. Similar to a BWR, the SCWR uses two feedwater lines made of carbon steel. However, BREI has determined that because of its high-density steam, the SCWR needs only two steam lines as opposed to four in a BWR of similar thermal power. This further adds to the economic strength of the SCWR concept. The steam lines can be constructed out of ferritic steels such as P91 and P92, which are currently used in supercritical fossil plant steam lines, although the latter would have to be included in Section III (nuclear components) of the ASME Code.

A pressure-suppression type containment with a condensation pool, essentially the same design as modern BWRs, was selected by Westinghouse. The dry and wet well volumes were calculated to limit the pressure build-up to typical BWR levels following a LOCA or a severe accident with core melting. The condensation pool water inventory was designed to provide ample margin for residual heat removal and meet the requirement that active safety systems are not needed during the first 24 hours following an initiating event resulting in a severe accident. The very conservative European Utility Requirements for mitigation of severe accidents were adopted in sizing the containment and a core catcher was added to the design. Despite this conservative approach the SCWR containment is somewhat smaller than that of an advanced BWR of similar thermal power, and thus significantly smaller on a per unit electric power basis. Following a critical review of the severe accident mitigation strategy, further reduction of the containment volumes will be explored in FY-04.

Thermal-hydraulic and thermal-nuclear coupled instabilities were investigated at ANL with a frequency-domain linear stability analysis code based on single-channel thermal hydraulics, one-dimensional fuel heat conduction, and point-kinetics models. The BWR stability criteria were adopted and it was found that the SCWR is stable against core-wide in-phase oscillations at normal operating power and flow conditions.

A critical review of the LWR abnormal events and their NRC classification has been performed by Westinghouse and INEEL with the SCWR application in mind. Four events were singled out that could be potentially troublesome: (i) loss of feedwater flow, which in the once-through direct-cycle SCWR coincides with the loss of core flow, (ii) turbine trip without steam bypass, which pressurizes the system and could result in significant positive reactivity insertion because of the low density of the SCWR coolant, (iii) loss of feedwater heating, which also results in the insertion of positive reactivity because of the lack of feedwater mixing with hotter coolant in the vessel, and (iv) large break in the feedwater lines, which, if unmitigated, results in coolant stagnation in the core and rapid overheating of the fuel. A preliminary analysis of these four key events was performed at INEEL with a modified version of the RELAP5 code. It was found that the SCWR behavior is relatively benign during the turbine trip without steam bypass, the loss of feedwater heating, and the large break in the feedwater lines. On the other hand, survival of the total loss of feedwater will likely require the use of a high-capacity high-pressure fast-acting auxiliary feedwater system. Design of such system will be a major challenge.

The reference SCWR system has a power conversion cycle that is very similar to a supercritical coal-fired plant, with the boiler replaced by the nuclear reactor. A conceptual study was performed by BREI to identify an optimal configuration for the goals of thermal efficiency maximization and capital cost minimization. The SCWR power conversion cycle uses a single-shaft turbine-generator, operating at reduced speed (1,800 rpm), with one high-pressure/intermediate-pressure (HPT/IPT) turbine unit and three low-pressure turbine (LPT) units with six flow paths, with a moisture separator reheater between the HPT/IPT and the LPTs, eight feedwater heaters, steam-turbine-driven feedwater pumps and natural draft cooling towers. The reference design generates 1,600 MWe with a thermal efficiency (net electric power to the grid / fission power) of 44.8% versus about 35% for LWRs under equivalent assumptions.

The feasibility of the 1,600-MWe turbine-generator was verified by BREI with various turbine vendors. A stage-by-stage model of the HPT/IPT and LPT was generated and demonstrated that the steam parameters at the turbine inlet, the steam speeds in the LPT exhaust annulus, the LPT blade lengths and moisture content are all within current standard ranges for steam turbines. Note that the overall physical size of the SCWR turbine generator is similar to a 1,300-1,500 MWe LWR turbine generator because the volumetric steam flow processed in the LPT is similar for both plants. All other major components including the feedwater heaters, pumps, cooling tower, steam lines, condenser, etc. have been sized and are either commercially available or within current design capabilities. Candidate materials for all components of the power conversion cycle have been identified by ORNL based on the fossil-plant experience.

A pre-conceptual design of the SCWR control system was also performed by BREI. The main characteristics affecting the design of the SCWR control system are the relatively low vessel water inventory, the nuclear/thermal-hydraulic coupling, the lack of level indication under supercritical conditions and the absence of recirculation flow. The main variables to be controlled include the reactor power, the core outlet temperature during supercritical pressure operation (e.g., full power operation), the reactor pressure, the reactor level during subcritical pressure operation (e.g., during start-up) and the feedwater flow. Then, assuming base-load operation, the recommended approach for the SCWR is one in which the control rods accomplish the primary control of the thermal power, the turbine control valve provides the control of the pressure, the feedwater flow (i.e., the feedwater pumps) provides the primary control of the outlet temperature, and the control of the coolant inventory in the vessel is accomplished by assuring that steam and feed flow are balanced while maintaining the correct core outlet temperature. Also, rather than an approach in which higher functions such as power or turbine valve control are in manual with lower level control loops in automatic, the use of an integrated control approach, one in which all functions are in automatic, is deemed preferable due to the SCWR's expected fast response to perturbations.

Start-up and shutdown procedures and related equipment for supercritical fossil plants were reviewed and their applicability to the SCWR plant was assessed. BREI determined that the use of a hybrid variable pressure start-up approach is preferable. This approach requires a start-up turbine bypass, a steam-water separator, drain valves, and recirculation pumps. The integrated control system will vary the pressure during start-up, but will do so in discrete steps with about three pressure set-points that will be established by the operator until supercritical operation is reached. The needed sequence and procedures for both start-up and shutdown of the SCWR plant were developed. The start-up procedures are similar to a LWR except for the transition to and from supercritical-pressure operation.

The issue of transport of coolant activation products to the balance of plant was also evaluated at INEEL. It was found that the ^{16}N activity in the SCWR steam is about twice that in the steam of a BWR with hydrogen water chemistry. However, a simple gamma attenuation model showed that this results in shielding requirements for the SCWR only up to 12% higher than for the BWR. Moreover, because of the higher SCWR electric power, the specific shielding costs ($/kWe) associated with ^{16}N are expected to be similar to or better than the BWR's.

In summary, the research work during the first year of the Generation-IV SCWR program has confirmed the basic assumptions contained in the Generation-IV Roadmap Report regarding the SCWR, and no new potential showstoppers have been found. The key feasibility issues for the SCWR remain the development of in-core materials and the demonstration of adequate safety. Dynamic instabilities appear to be less of a concern.

Detailed technical information on the experiments and analyses briefly discussed in this report can be found in the annual progress reports produced by the performing organizations, which are available from the SCWR Product Manager upon request.

Table of Contents

1. Introduction and Costs

The supercritical water-cooled reactor (SCWR) is one of the six reactor technologies selected for research and development (R&D) under the Generation-IV program. 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, steam separators and dryers 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 is 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 [GIF 2002, Kataoka et al. 2002, Spinks et al. 2002, Squarer et al. 2002], and will not be repeated here.

In the U.S. the Generation-IV SCWR program is led by the INEEL and 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[1].

In FY-03 the Generation-IV SCWR program in the U.S. comprised six tasks involving six organizations, i.e., INEEL, ANL, ORNL, the Westinghouse Electric Company (including the BWR Engineering group in Sweden), Burns & Roe Enterprises Inc. (BREI) and MIT. The total budget was $438,000 with the cost breakdown shown in Table I. Excellent technical progress has been made, and all the milestones indicated in the program plan were met on time. The key technical findings are explained in Section 2, program management activities are reported in Section 3, while other non-Generation-IV SCWR activities in the U.S. (including NERI and I-NERI projects) are briefly discussed in Section 4.

Table I. Task and cost breakdown for the U.S. Generation-IV SCWR program in FY-03.

Task	Organization	Budget ($k)	Actuals* ($k)
Program Management	INEEL	75	75
Balance of plant design, control and start-up	BREI (INEEL subcontract)	50	50
Stability analysis	ANL	53	53
Containment and safety systems design	Westinghouse	160	101
Materials survey	ORNL	50	50
Corrosion testing of candidate materials	MIT (INEEL subcontract)	50	50
Total		**438**	**379**

* As of 9/29/03

All the technical details of the analyses and experiments mentioned in this report can be found in the following publications, which are available from the SCWR Product Manager upon request:

[1] Similar assumptions are adopted in other countries with the notable exception of Canada were the focus is on a light-water-cooled, heavy-water-moderated SCWR concept.

1) Balance of plant, reactor control and start-up (BREI)

Burns & Roe Enterprises Inc., *Supercritical Water Reactor (SCWR), Study of Power Conversion Cycle, Control Strategy and Start-up Procedures*, September 2003.

2) Stability analysis (ANL)

W. S. Yang, N. Zavaljevski, *Preliminary Investigation of Power-Flow Instabilities of Supercritical Water Reactor*, Argonne National Laboratory, September 2003.

W. S. Yang, N. Zavaljevski, "Preliminary Stability Analysis for Supercritical Water Reactor", Paper 87886, *Proceedings of Global 2003*, New Orleans, November 16-20, 2003.

3) Containment and safety systems design (Westinghouse)

N. O. Jonsson, U. Bredolt, T. A. Dolck, A. Johanson, T. Ohlin, L. Oriani, L. Conway, *SCWR - Design Review and Design of Safety Systems and Containment – Status, September 2003*, SE-03-044 (Rev. 0), September 2003.

4) Materials survey (ORNL, INEEL)

J. Buongiorno, W. Corwin, P. E. MacDonald, L. Mansur, R. Nanstad, R. Swindeman, A. Rowcliffe, G. Was, D. Wilson, I. Wright, *Supercritical Water Reactor (SCWR), Survey of Materials Experience and R&D Needs to Assess Viability*, INEEL/EXT-03-00693 (Rev. 1), Idaho National Engineering and Environmental Laboratory, September 30, 2003.

5) Corrosion testing of candidate materials (MIT, University of Michigan)

J. McKinley, S. Teysseyre, G. S. Was, D. B. Mitton, H. Kim, J-K Kim, and R. M. Latanision, "Corrosion and Stress Corrosion Cracking of Austenitic Alloys in Supercritical Water", Paper 1027, *Proceedings of GENES4/ANP2003*, Kyoto, JAPAN Paper 1027, Sep. 15-19, 2003.

2. Technical Progress in FY-03

2.1 General Plant Characteristics

The reference SCWR design for the U.S. program is a direct cycle system operating at 25.0 MPa with core inlet and outlet temperatures of 280 and 500°C, respectively. 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 reference power, efficiency, pressure, and coolant flow rate and temperatures are listed in Table II. Figure 1 is a sketch of the reactor pressure vessel and internals showing the coolant flow paths. The components limiting the power rating of the SCWR are the turbine and the reactor pressure vessel. The feasibility of the pressure vessel design and the turbine design for the selected power level is discussed in Sections 2.2 and 2.7, respectively.

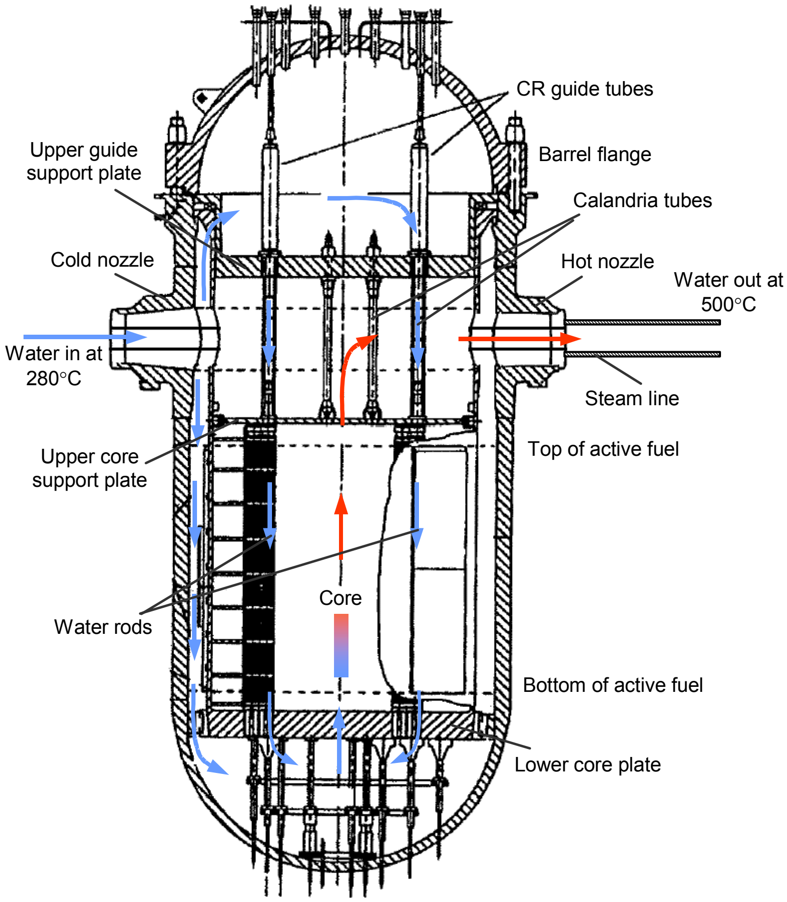


Figure 1. The SCWR reactor pressure vessel.

Table II. 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

2.2 Reactor Pressure Vessel (RPV)

The key characteristics of the SCWR vessel are listed in Table III, and a two-dimensional isometric view is shown in Figure 2. This vessel design is similar to a typical large-size PWR vessel design with no major penetrations through the lower head. However the thickness is significantly larger due to the higher operating pressure. The reactor flow path is designed to keep the whole RPV at 280°C (the feedwater temperature), which requires the use of a thermal sleeve for the outlet nozzle. Then typical state-of-the-art LWR materials can be used, i.e., SA 508 Grade 3 Class 1 for the shell and head, clad with a weld overlay of 308 stainless steel; Alloy 82 can be used for welding at nozzles and attachments. The use of standard LWR materials for the RPV is a major economic advantage for the SCWR compared with other Generation-IV concepts such as the gas-cooled reactors which may require the use of advanced alloys operating at much higher temperatures.

Table III. The SCWR RPV parameters.

Parameter	Value
Type	PWR with top CRDs
Height	12.40 m
Material	SA-508
Operating/design press.	25.0/27.5 MPa
Operating/design temp.	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)	$<10^{20}$ n/cm^2

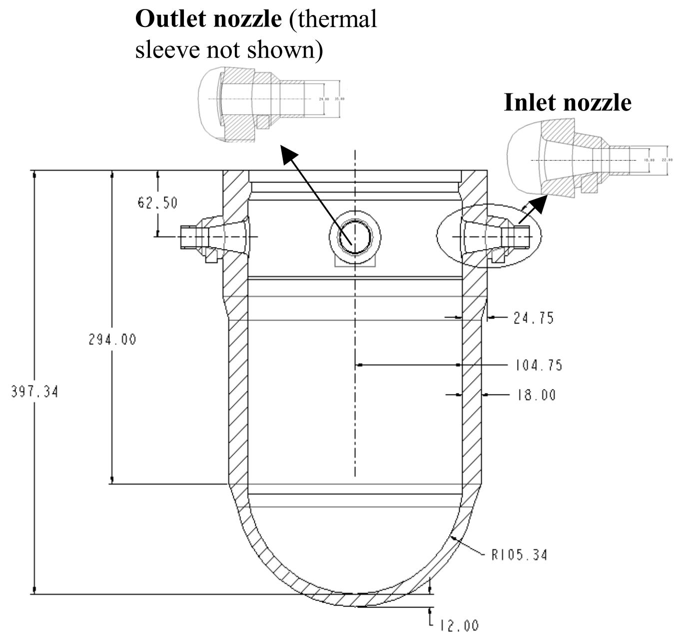


Figure 2. Dimensions of the SCWR RPV showing the inlet and outlet nozzles (lengths in inches).

The SCWR RPV was sized by Westinghouse to meet the requirements of the ASME code, Section III, Class I, NB-3324, and its structural design was verified both at Westinghouse and INEEL with detailed three-dimensional finite-element analyses including the effect of penetrations, vessel weight and thermal stresses. The RPV is vertically supported below all four nozzles.

The expected radiation damage to the vessel over the 60-yr lifetime is within typical PWR range due to a similar downcomer width and somewhat lower power density. Nevertheless radiation embrittlement issues will be minimized by controlling the use of sensitizing materials (Cu, P) in the weld regions and by fabricating a single ring forging for the active core region to avoid the need for circumferential welds in that region. Also, a surveillance program will be implemented to monitor the evolution of the thick sections of the vessel.

Note that the SCWR RPV is beyond current manufacturing capabilities in the U.S. The beltline-region ring forging is 4.3 m corresponding to the active core height. This slightly exceeds the height of the largest SA 508 forged rings made to date (about 4 m) by Japan Steel Works (JSW) for the ABWR. However, JSW has indicated that they should be able to build longer and thicker forgings with some modest changes in their equipment. JSW is limited by the total weight of any given forging or about 600 t, which is much higher than the weight of the 4.3-m long SCWR RPV ring forging (about 285 t).

ORNL has assessed the possibility of using advanced (higher-strength) materials to reduce the thickness and weight of the SCWR RPV. These include A508 Grade 4N Class 1 and a developmental steel, 3Cr-3WV. Use of these steels would allow for more than a 30% reduction in shell thickness, which could significantly reduce the fabrication costs, assuming a material cost not much higher than SA-508's. However, significant additional mechanical property data would be needed for these materials to allow for their inclusion in the ASME code, and irradiation effects data for all relevant mechanical properties would be required for licensing.

Outstanding issues for the SCWR RPV include (i) the design of the thermal sleeve, which has not been performed yet, but is key to the RPV feasibility, (ii) formal confirmation of the manufacturing capabilities for the beltline ring forging at JSW or other manufacturers, (iii) maintenance of through-thickness mechanical and chemical properties during fabrication, and (iv) monitoring of flaw density in the very thick shell, which will be a challenge especially at weld locations.

2.3 Core and Fuel Assembly Design and Materials Selection

The reference SCWR core design is shown in Figure 3. The relevant dimensions are listed in Table IV. The core has 145 assemblies with an equivalent diameter of about 3.9 meters. The average power density is about 70 kW/L (or 30% higher than BWRs and 35% lower than PWRs) with a total target power peaking factor of about 2.0. The average and peak linear heat generation rates are similar to typical LWR values. The estimated core pressure drop is also comparable with typical LWR pressure drops and inlet orifices are used to adjust the flow to each assembly based on its expected power.

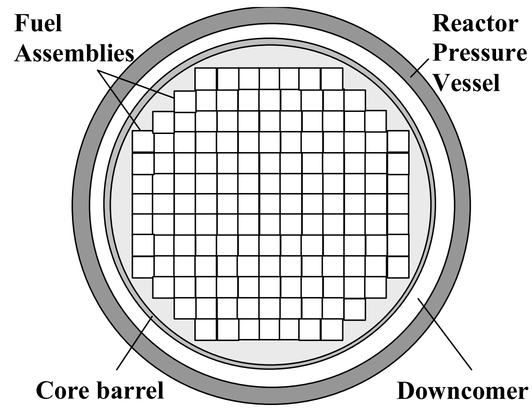


Figure 3. Sketch of the reference SCWR core.

Table IV. Reference reactor core design for the U.S. Generation-IV SCWR.

Parameter	Value
Number of fuel assemblies	145
Equivalent diameter	3.93 m
Core barrel ID/OD	4.3/4.4 m
Axial/Radial/Local/Total Peaking Factor	1.4/1.3/1.1/2.0 (best estimate) 1.4/1.4/1.2/2.35 (safety analysis)
Average power density	69.4 kW/L
Average linear power	19.2 kW/m
Peak linear power at steady-state conditions	39 kW/m
Core pressure drop	0.15 MPa
Water rod flow	1659 kg/s (90% of nominal flow rate)

The reference SCWR fuel assembly design is shown in Figure 4 and the relevant dimensions are listed in Table V. The fuel assembly has square water rods and an external duct. Analyses performed at the INEEL have shown that it may be necessary to insulate the water moderator boxes to retain a sufficient moderator density. The appropriate insulating material has not yet been determined. Figure 10 also shows the control rods inside 16 water moderator boxes, including the control rod thimbles. However, the control rod worth calculations are not complete and it may be desirable to change the number and/or size of the control elements, or it may be desirable to change the locations of the control elements. Also, it is assumed that there is one instrumentation tube in each assembly at the center fuel rod location. The number of the dimensions are tentative including the fuel bundle wall thickness and the inter-assembly gap size, and the fuel pin spacers have yet to be designed.

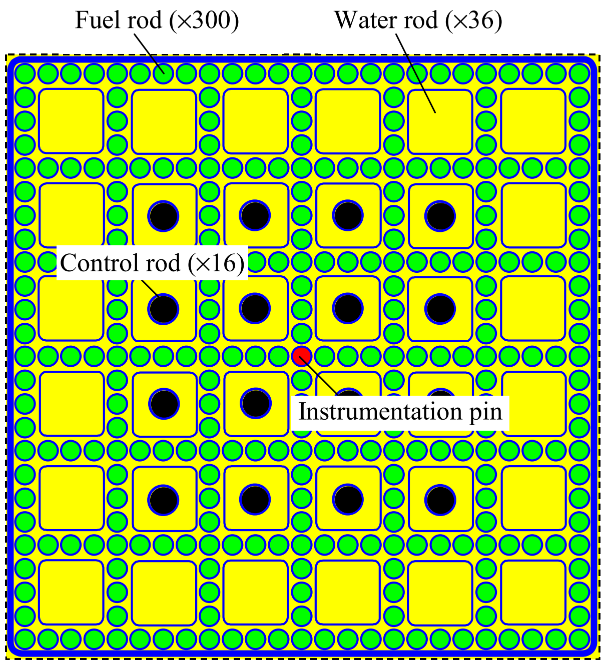


Figure 4. The SCWR fuel assembly with water rods.

Table V. 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
Control rod material	B_4C for scram, Ag-In-Cd for control
Number of spacer grids	14 (preliminary estimate)
Assembly duct thickness	3 mm (plus insulation if needed)
Assembly side	286 mm
Inter-assembly gap	2 mm
Assembly pitch	288 mm

The reference fuel pin dimensions are listed in Table VI. The fuel pin dimensions are typical of 17×17 PWR fuel assembly pins, with the exception of the plenum length and fill pressure. Thermo-mechanical analysis of the SCWR fuel pin were performed at the INEEL with the FRAPCON code have shown that a higher fill pressure is needed to prevent buckling at beginning of life and a longer fission gas plenum is needed to limit the internal pressure at end of life. Also, to enhance coolant velocity and heat transfer, the fuel pin pitch is considerably smaller than the pitch used in LWRs. The U-235 enrichment, the Gd_2O_3 loading, and the fuel burnup are typical of the values used in high burnup LWR fuel, although their distribution within the fuel pin, within the fuel assembly and throughout the core are yet to be determined.

Table VI. Reference fuel pin design 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
Fuel pellet outside diameter	8.78 mm
Fuel composition	UO_2, 95% TD
Fuel enrichment	5% wt. average
Target average burnup at discharge	45,000 MWD/t
Burnable poisons	Gd_2O_3 (distribution TBD)
Heated length	4.27 m
Fission gas plenum length	0.6 m
Total fuel pin height	4.87 m
Fill gas pressure at room temperature	6.0 MPa

Candidate materials have been identified by the ORNL materials experts for all the components of the fuel assembly. Table VII lists these components together with summaries of the anticipated irradiation conditions and mechanical loads for normal operating conditions, as well as the temperature excursions expected for abnormal conditions. Also listed are typical materials for similar components in currently operating PWRs and BWRs. The last two columns of the table give recommendations for potential candidate materials for the SCWR, together with brief notes to further explain or augment other entries in the table. The structural materials recommended for these components are primarily ferritic-martensitic steels (e.g., T91, A-21, NF616, HCM12A), and low swelling variants of the austenitic stainless steels (e.g., D-9, PNC). Among the more advanced materials oxide-dispersion strengthened ferritic steels (e.g., MA-957) and ceramic composites (e.g., SiC-SiC) should also be explored given their potential for superior high-temperature strength. Many of these materials have been selected based on satisfactory unirradiated properties and/or proven performance under irradiation. A more thorough discussion of the material selection and a complete materials development program can be found in the materials survey report [Buongiorno et al. 2003a]. Possible insulating materials for the water rods will be investigated in FY-04.

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

Component	Normal Conditions			Abnormal Conditions	Current LWR Materials		Candidate SCWR Materials	Notes
	Temperature [1]	Peak Dose [2]	Loads [3]	Temperature [4]	PWR	BWR		
Fuel cladding	280-620 ºC	15 dpa	ΔP across cladding (compressive stresses up to 100 MPa), grid-cladding and fuel-cladding interactions	Up to 840 °C for <30 sec	Zircaloy 4	Zircaloy 2	Fe-Ms, Low-swelling 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-swelling 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-swelling S.S., SiC-SiC	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-swelling S.S., SiC-SiC	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. Traditional low-swelling stainless steels, e.g., D-9 (14.5Cr-14.5Ni, 2Mo, Ti stab), PNC ~D-9 mod w/P).					

2.4 Vessel Internals Design and Materials Selection

The important reactor pressure vessel 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 1. All the reactor pressure vessel internals components will be designed for periodic replacement so that very high fluence loadings will not need to be considered.

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 reactor pressure vessel internals, including the core barrel (or possibly the core former, depending on the design details), the upper guide support plate, the calandria tubes, and the reactor pressure vessel hot nozzle sleeve, will be in contact with coolant at the inlet temperature of 280°C on one side and the hot outlet coolant at a temperature of 500°C on the other side. Preliminary stress analyses performed at Westinghouse indicate that metal wall designs that are similar to those currently used in LWRs for those components cannot be used. Such a high temperature drop across those walls will cause the thermal stresses and deformations to be too large and/or cause too much heat to be transferred across the walls. For example, a simplified thermal stress analysis of the upper guide support plate was performed using a temperature difference of 220°C (396°F) and the Pro/Mechanica software. The result was that much of the structure will exceed the 3 Sm Primary + Secondary stress limit of Subsection NB of the ASME code as shown in Figure 5. Resolution of these issues may require new design features including special materials, insulation layers, and/or use of an insulating layer between double walls. Possible insulating materials for the vessel internals will be explored in FY-04. Some other reactor pressure vessel internals components, such as the upper core support plate, will be exposed to the outlet coolant at a temperature of about 500°C on all sides, and will not require insulation.

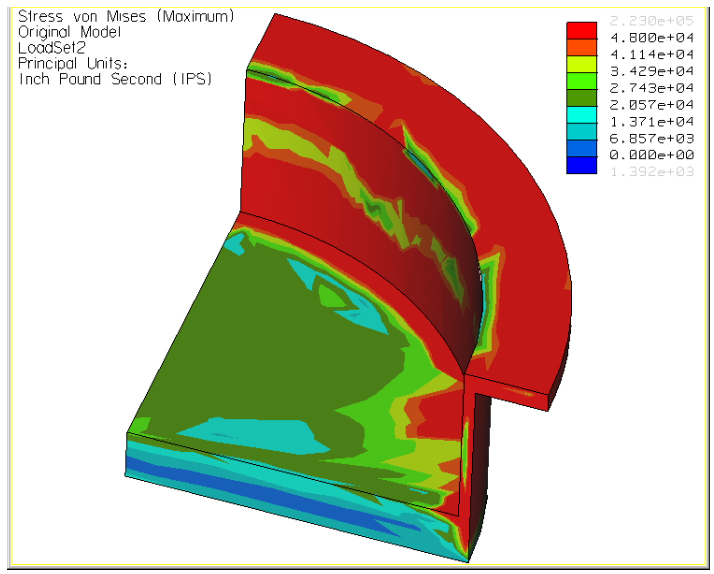


Figure 5. Results of the preliminary thermal stress analysis of the upper guide support plate.

The size and shape of most of the reactor pressure vessel internals discussed above should be similar to comparable components in a large Westinghouse designed PWR. However, it should be noted that the design of the calandria tubes that guide the flow of the moderator water through the hot region above the core and guide the control rods is not complete. There is a need to minimize the heat transfer surface area, and one way to do that is to combine the outside water moderator boxes into one channel in the region above the core.

Table VIII. 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	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	May want to use the same material as for the UGS, UCS, and calandria tubes
Core Former	~280-600 ºC	~20 dpa [4]	Significant hydraulic. High thermal.	700ºC	304 S.S.	N/A	Fe-Ms, Low-Swelling 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-Swelling 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 [5]			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 are320-370 °C
2. Design estimates for 60y
3. All reactor internals will be subject to seismic and pipe break loads
4. Assuming the core former is replaced twice over the 60-yr lifetime of the plant
5. ~ 50 dpa for baffle bolts and formers in PWRs. Here it is assumed that fasteners will not be used in the high flux regions.

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

Materials recommendations for all vessel internals are reported in Table VIII. Again ferritic steels and low-swelling stainless steels are recommended for the components more exposed to the neutron flux, while high-strength stainless steels and nickel-based alloys can be used in regions where low radiation damage is expected.

In FY-03 two traditional austenitic steels (304L and 316L) were tested at MIT and the University of Michigan for corrosion and stress-corrosion cracking (SCC) susceptibility in supercritical water. Although it is recognized that these alloys are not particularly promising because of their relatively poor irradiation stability and high-temperature strength, a large database for corrosion and SCC is available at LWR conditions, and it was deemed useful to compare that known behavior with their behavior in supercritical water. It was found that both alloys are susceptible to SCC (316L less so than 304L) in both deaerated and non-deaerated high-temperature (>400°C) supercritical water [McKinley et al. 2003]. Thus, these alloys cannot be used for high-temperature components in the SCWR. However, they could be used for components operating in the 280-350°C range (e.g., the lower core plate, the control rod guide tubes), given their satisfactory behavior in deaerated water at these temperatures.

2.5 Reactor Coolant System

The reactor coolant system discussed in this section comprises (i) the feedwater lines from the isolation valves to the RPV, and (ii) the steam lines from the RPV up to the second set of main steam isolation valves outside the containment. The balance of plant is discussed in Section 2.7. In the ASME Code terminology the feedwater and steam lines within the containment are Class 1 components, and any break they might experience is considered a loss-of-coolant accident (LOCA). The SCWR reactor coolant system was designed by BREI and has two feedwater lines and two steam lines (as opposed to four in a BWR of similar thermal power). The main parameters of the SCWR reactor coolant system are reported in Table IX.

Table IX. 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
	Reference materials	SA-106 Grade C (carbon steel)
Steam lines	Number	2
	Operating temperature	500°C
	Operating/design pressure	25/27.5 MPa
	OD/thickness	470 mm / 51 mm
	Reference material	P91 (9Cr-1Mo) or P92 (9Cr-2W)

The SCWR feedwater lines can likely use standard LWR materials such as carbon steels, perhaps clad with stainless steel if thus dictated by the water chemistry. The selection of suitable materials for the steam lines is more problematic. While the SCWR steam lines operate at temperatures and pressures that are well within the supercritical fossil plant experience, the direct application of fossil plant materials is not straightforward because of the ASME Code regulations. For example, Alloy 91 (P91) is used for fossil plant steam lines and is also approved for use in Subsection NH (high-temperature applications) of the ASME Section III (nuclear components), but its lifetime at temperature is limited by the Code to only

34 years vs. the intended 60 years of life in the SCWR plant. So either the steam lines are replaced after 34 years or the allowable lifetime for P91 in Subsection NH must be extended to 60 years. There are also some alternate (stronger and more corrosion resistant) materials that could be considered. For example, P92 (9Cr-2W) is an advanced ferritic steel also used in fossil-fired supercritical plant steam lines, and meets the requirements of B31.1 (non-nuclear applications), but would have to be qualified for Section III.

2.6 Containment

The SCWR containment was designed by Westinghouse [Jonsson et al. 2003] and is a pressure-suppression type containment with a condensation pool (essentially the same design as modern BWRs). Key containment parameters are listed in Table X. A three-dimensional isometric sketch of the SCWR containment is shown in Figure 6 and an axial cross-section with dimensions is shown in Figure 7. The dry and wet well volumes were calculated to limit the pressure build-up to typical BWR levels following a LOCA or a severe accident with core melting (hydrogen generation from cladding oxidation was considered in the calculations). The concrete floors were designed to withstand such loads. The condensation pool water inventory provides ample margin for residual heat removal and meets the requirement that active safety systems are not needed during the first 24 hours following an initiating event resulting in a severe accident. The blow-down pipes or vents are placed in the outer cylindrical walls due to lack of space in the inner cylindrical walls.

Table X. SCWR containment parameters.

Parameter	Value
Dry well volume	5000 m^3
Wet well gas volume	3300 m^3
Wet well condensation pool volume	5640 m^3
Blow-down area	18 m^2 (~60 vents)
Dry well maximum pressure	510 kPa
Wet well maximum pressure	470 kPa
Dry to wet well maximum pressure difference	300 kPa
Dry well temperature local (short time)	500°C
Dry well temperature global (short time)	350°C
Dry well temperature global (long time)	150°C
Wet well gas temperature	100°C
Condensation pool temperature	<100°C

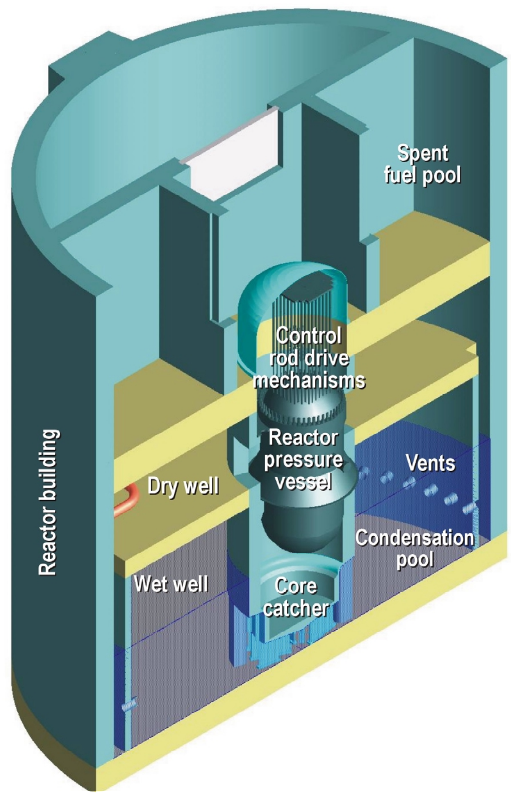


Figure 6. SCWR pressure suppression pool type containment.

Compared to the advanced BWR containment designs, the SCWR containment drywell can be reduced because:

- The SCWR has only two steam and feedwater lines.
- The SCWR has a smaller diameter of the pressure vessel.
- The control rods enter the reactor pressure vessel from the top. Also, there are less control rod drive installations needed and fewer areas needed for transportation of equipment. Also, installations for control rod drive maintenance are not needed below the pressure vessel.
- There are no internal recirculation pumps.

On the other hand, the SCWR containment drywell volume is increased because of the high temperature fluid moving from the reactor to the turbine, since additional cooling and thermal expansion space are needed. Also, the concrete must accommodate higher temperatures during an accident. Furthermore the SCWR containment is lower because the pressure vessel is lower. However, this will tend to increase the diameter of the containment and will also lead to less space for connections and floorings. When all these effects are accounted for, the SCWR containment ends up being somewhat smaller than that of an advanced BWR of similar thermal power, and thus significantly smaller on a per unit electric power basis.

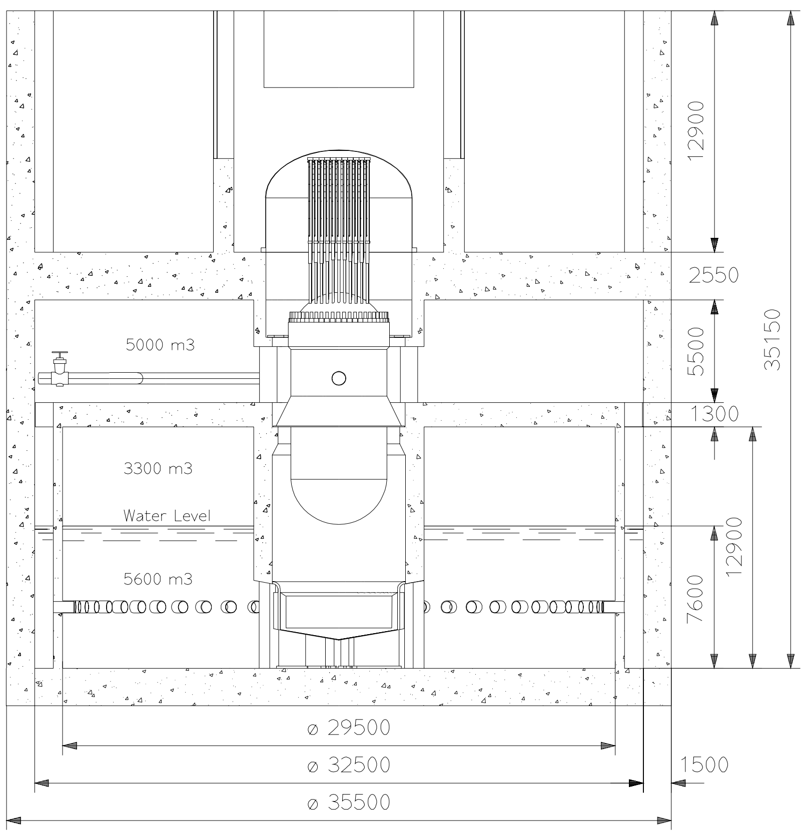


Figure 7. SCWR pressure suppression containment building.

Our interpretation of the Generation-IV goal of superior safety is that because the potential for core damage in a SCWR is similar to traditional LWRs, enhanced safety is only possible if one can claim that the offsite consequences of a core damage accident are negligible. The European Utility Requirements statements regarding severe accidents and mitigation of their effects were adopted: "Core debris cooling. This can be achieved via a solidly founded technical demonstration for either in-vessel debris cooling or ex-vessel debris cooling." The current SCWR design includes a core catcher under the reactor pressure vessel, thus achieving ex-vessel retention. However, based on the power rating and vessel size of the SCWR, an alternative solution featuring in-vessel core debris cooling should also be possible. As already mentioned, the condensation pool is sized to provide a sufficient heat sink for decay heat in case of severe accident. This approach leads to a larger containment size, but simplifies the design of safety and mitigation systems. Other alternatives should be possible to provide the same grace period following a severe accident, e.g., a passive containment cooling system. It is difficult to judge the best solution at this point of the design, and it was, therefore, decided to proceed with this reference solution. Based on a critical review of the severe accident mitigation strategy, further reduction of the containment volumes will be explored in FY-04.

Outstanding issues for the SCWR containment include an evaluation of the consequences of the high local temperatures following blow-down and verification that the safety systems can be accommodated within the containment.

2.7 Dynamic Analysis of the SCWR Reference Plant

2.7.1 Stability Analysis

Consistent with the U.S. NRC approach to BWR licensing, the licensing of SCWRs will probably require, at a minimum, demonstration of the ability to predict the onset of instabilities. This can be done by means of a 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.

Thermal-hydraulic and thermal-nuclear coupled instabilities were investigated at ANL[2] [Yang and Zavaljevski 2003]. A frequency-domain linear stability analysis code was developed based on single-channel thermal hydraulics, one-dimensional fuel heat conduction, and point-kinetics models. The reactivity feedback coefficients were calculated with the WIMS-8 lattice code. Following the standard approach for BWR stability analysis, the system stability was estimated using the decay ratio, which is determined by searching the dominant root of the system characteristic equation directly in the complex plane. Preliminary stability analyses were performed for the reference SCWR concept. It was found that the core-wide in-phase oscillations would decay quickly at normal operating power and flow conditions. The estimated hot-channel decay ratios for thermal-hydraulic and thermal-nuclear coupled instabilities are 0.20 and 0.007, respectively, which are well below the limits traditionally imposed for BWR stability (0.5 and 0.25 for thermal and thermal-nuclear oscillations, respectively). Sensitivity studies showed that increasing the pressure loss coefficient of the inlet orifice increases the system stability, while increasing the coolant density feedback coefficient decreases it. Also the effect of the axial power profile was investigated and it was found that, compared to the cosine shape axial power profile, a uniform power profile improves stability, but a bottom-skewed profile decreases it slightly.

2.7.2 Preliminary Analysis of Key Transients and Accidents

Westinghouse and INEEL performed a critical review of the LWR abnormal events and their NRC classification with respect to the SCWR application. The objective was to identify potentially troublesome events on which to focus the R&D attention as early in the program as possible. The following four events were singled out:

[2] Dr. Pradip Saha at MIT is also conducting SCWR stability analysis with internal funding. His results are consistent with ANL's.

- Total loss of feedwater flow. Because the SCWR is a once-through direct cycle without coolant recirculation, a loss of feedwater flow immediately causes a loss of core flow and results in rapid undercooling of the core.
- Turbine trip without steam bypass. The average coolant density is low in the SCWR core and pressurization events (such as the turbine trip without stem bypass or the accidental closure of the main steam isolation valves) result in significant positive reactivity insertion and increase in reactor power.
- Loss of feedwater heating. When a feedwater heater is lost, relatively cold water enters the core resulting in the insertion of positive reactivity. The difference between the behavior of a SCWR and a BWR is that the effect is expected to be more pronounced, because the feedwater is not mixed with hotter water before entering the core.
- Large break in the feedwater lines (or cold-leg large-break LOCA). Because the SCWR coolant path is once-through without recirculation in the vessel, an unmitigated large break in the feedwater lines results in coolant stagnation in the core and rapid overheating of the fuel.

Note that the first three events are classified by the U.S. NRC as moderate-frequency (Condition II) events for LWRs and must not result in any significant damage to the fuel, while the large-break LOCA is classified as a rare event (Condition IV) and limited damage to the fuel is permitted.

Because at supercritical conditions the boiling crisis does not occur, the traditional CHF criterion to assess the margin to failure cannot be used. Maximum allowable cladding temperatures are specified instead for both transients (Condition II events) and accidents (Condition III and IV events). Note that a cladding temperature criterion (<1205°C, <2200°F) is also used for LWRs in the analysis of large-break LOCAs, and thus is adopted for the SCWR accidents as well[3]. The limit for transients is assumed to be 840°C, which is a reasonable value, but will have to be verified with fuel performance codes. The following limits are also assumed for the SCWR fuel, and are identical to those used in BWRs: no centerline melting under transient overpower, and radial-averaged fuel enthalpy (at any location in the core) below 0.711 kJ/g (170 cal/g) during transients and below 1.17 kJ/g (280 cal/g) during accidents.

A preliminary analysis of the SCWR response to the four events mentioned above was performed at the INEEL. The purpose of the analysis was to characterize the time constants of the system so that the required response times and capacities for various safety systems could be determined. The analysis was performed using a modified version of RELAP5-3D, specifically improved to support analysis of the SCWR. The improvements included changes of the water properties interpolations around the critical point, modification of the solution scheme in the supercritical region, and addition of heat transfer and wall friction correlations applicable to supercritical conditions. The RELAP5 model includes the average channel, the hot channel (with radial, axial and local power peaking factors of 1.4, 1.4 and 1.2, respectively, but without the other hot channel factors) and three core bypass paths; uses inlet orifices to minimize power/flow mismatches and boundary conditions to represent the feedwater and main steam systems; and calculates the transient reactor power with a best-estimate point kinetics model using reactivity feedback generated with the MCNP-4B code.

The effect of the main feedwater (MFW) pumps coastdown time, scram delay time and auxiliary feedwater (AFW) flow rate was evaluated for the total loss of feedwater. Figure 8 shows that the SCWR meets the transient peak cladding temperature criterion with the following assumptions: a MFW pump coastdown of 5 s, AFW flow is initiated at 4.25 s and corresponds to 15% of the initial MFW flow, the reactor scram signal is generated at 0.5 s, triggered by a 10% reduction in MFW flow, the control rods

[3] However, note that this criterion may be conservative for the SCWR system, which does not use zirconium-based cladding.

begin moving 0.8 s later and are fully inserted 2.5 s later, the reactor pressure is assumed to remain constant due to the operation of turbine bypass valves. Thus, the SCWR will likely need an AFW system that is fast acting and of relatively high capacity; this will be a significant design challenge.

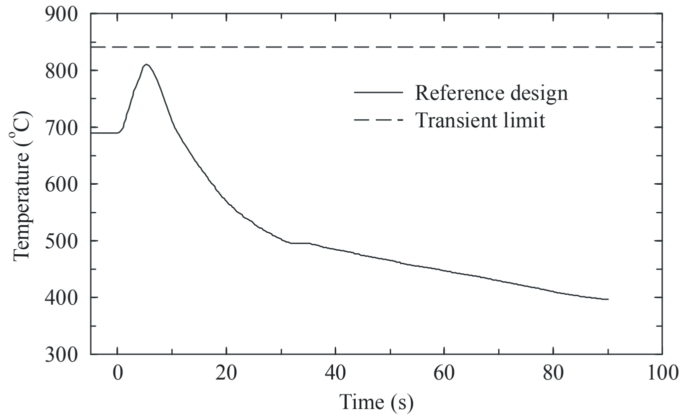


Figure 8. Peak cladding temperature during the loss of MFW flow event for the SCWR.

The analysis also showed that the turbine trip without steam bypass is fairly benign because the high-capacity steam relief valves open quickly and prevent over-pressurization of the system, i.e., the inherent behavior of the SCWR is very similar to a BWR. However, the reactivity insertion is not as high as in the BWR because most of the moderation in the SCWR core is obtained from liquid coolant (in the water rods), which is not affected by pressurization. As a result the fuel is not overheated. The reactor scram, triggered by the turbine trip, quickly terminates the event.

The loss of feedwater heating is also benign as shown in Figure 9. The cladding and the fuel (not shown) are not overheated because of the lower temperature coolant and the Doppler feedback effect, respectively. The assumptions are as follows: the event is initiated by a 30°C step decrease in feedwater temperature (corresponding to the loss of the last heater on the high-pressure feedwater train), the MFW mass flow is held constant during the transient, scram is not assumed, and the turbine bypass valves are assumed to hold the reactor pressure constant.

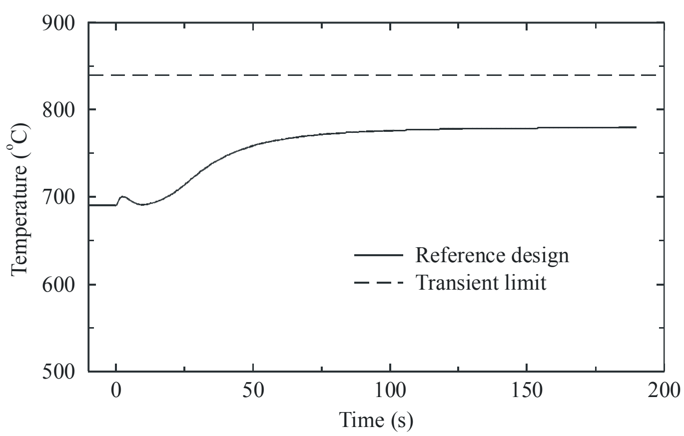


Figure 9. The effect of a 30°C step reduction in MFW temperature on the peak cladding temperature.

As far as the cold-leg large-break LOCA is concerned, the calculations were performed without emergency core coolant and automatic depressurization systems to provide an indication of the time available for these systems to operate. A 100% feedwater line break is assumed. Scram is not needed as the reactivity feedback due to the decrease in moderator density is able to quickly shut down the reactor. No explicit modeling of the RPV-containment coupling is provided at this stage, so the pressure downstream of the break is set to atmospheric pressure. The main feedwater flow in the other feedwater line is ramped linearly to zero over 5 s. Check valves in the steam lines close quickly. The results are shown in Figure 10. The SCWR reaches the accident limit of 1205°C in about 25 s. As in other LWRs, this will require the definition of both a high-pressure (active or passive) injection system for initial mitigation plus a low-pressure (active or passive, with a depressurization plus gravity injection solution).

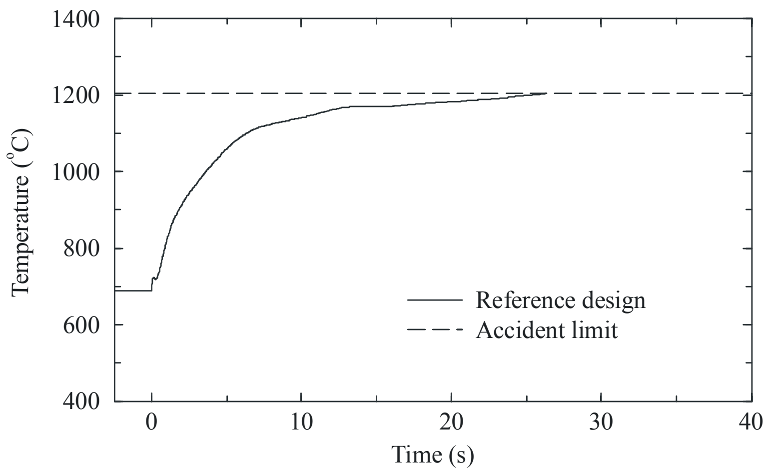


Figure 10. The effect of a 100% feedwater line break on the peak cladding temperature.

2.8 Balance of Plant (BoP)

The reference SCWR system has a power conversion cycle that is very similar to a supercritical coal-fired plant, with the boiler replaced by the nuclear reactor. BREI has performed a conceptual study of the power conversion cycle for the SCWR to identify an optimal configuration for the goals of thermal efficiency and electric power output maximization and capital cost minimization. Particular attention was also given to ensure that all components are either commercially available or within current design capabilities. The following trade-offs affecting the goals were considered: full vs. reduced speed of the turbine-generator module, single-shaft vs. multi-shaft arrangement of the turbine-generator module, steam-turbine-driven vs. motor-driven feedwater pumps. A schematic of the SCWR power conversion cycle is shown in Figure 11, while the operating conditions are reported in Table XI. The following design choices should be noted:

- Reduced rotation speed, 1800 rpm
- Single-shaft turbine-generator
- One high-pressure/intermediate-pressure turbine (HPT/IPT) unit and three low-pressure turbine (LPT) units with six flow paths
- Moisture separator reheater between the HPT/IPT and the LPTs
- Eight feedwater heaters raising the feedwater temperature to 280°C
- Steam-turbine-driven feedwater pumps operating at about 190°C
- Heat rejection in natural draft cooling towers

The reference design generates 1,600 MWe with a thermal efficiency (net electric power to the grid / fission power) of 44.8%. BREI has also sized the feedwater heaters, pumps, cooling tower, steam lines, condenser, etc. Note that, due to the higher steam density, only two small steam lines are needed for this large size SCWR vs. four lines for an LWR of comparable power, which further adds to the capital cost savings of the SCWR.

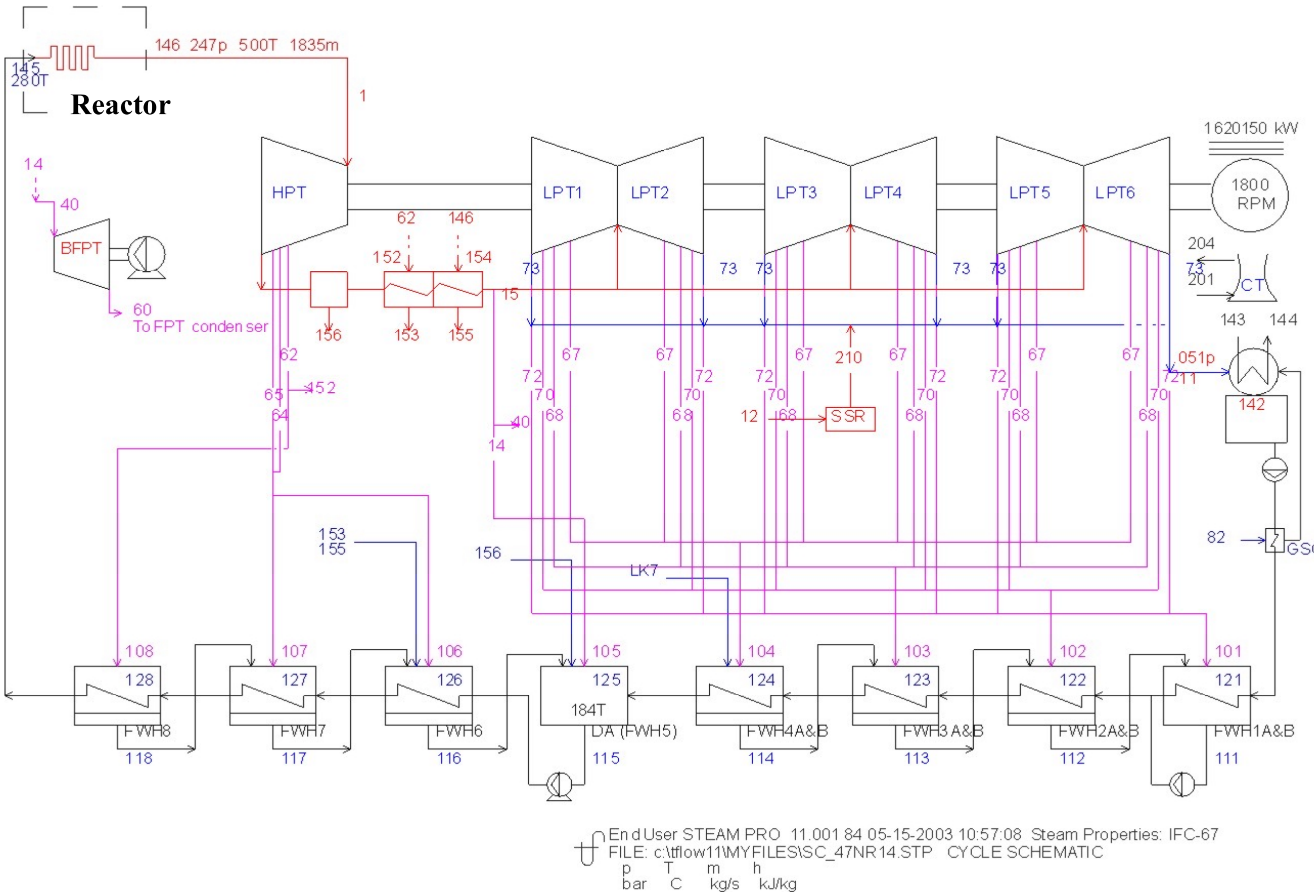


Figure 11. Schematic of the SCWR power conversion cycle (HPT = high pressure turbine, LPT = low pressure turbine, FWH = feedwater heater).

Table XI. List of pressures, temperatures, mass flow, and enthalpy at the numbered locations of Figure 11.

Stream	p [bar]	T [C]	T [kg/s]	h [kJ/kg]
1 Throttle or initial condition outside ST	235	494	1722.47	3167.3
6 PIPT ahead of intercept valve	12	188	1130.63	2773.7
11 Condenser (LPT exhaust	0.05	33.1	782.36	2290.3
12 SSR Inlet	1.24	105.8	0.94	2616.5
14 After 2nd RH	12	363	149.69	3182.2
15 LPT Crossover	12	363	982.07	3182.2
40 Inlet stream of FPT	11.43	361.4	96.15	3179.9
60 Extr1 (or exh if only 1 group) of FPT	0.07	38.7	96.15	2410.4
62 Add / extr of ST group 2	70	313.3	265.4	2893
64 Add / extr of ST group 4	45	259.4	127.38	2805.3
65 Add / extr of ST group 5	23	219.6	75.74	2684.8
67 Add / extr of ST group 7	5.4	264.2	13.39	2989.3
68 Add / extr of ST group 8	2.5	179.2	6.82	2825.1
70 Add / extr of ST group 10	0.6	86	9.84	2585.1
72 Add / extr of ST group 12	0.13	51.1	3.33	2382.1
73 Add / extr of ST group 13	0.05	33.1	130.3	2290.3
82 Stream to GSC 0.83	0.83	N/A	0.38	2616.5
101 Heating steam at FWH1	0.12	49.5	19.96	2379.8
102 Heating steam at FWH2	0.58	85	59.06	2582.7
103 Heating steam at FWH3	2.4	177.8	40.94	2822.8
104 Heating steam at FWH4	5.18	262.8	80.32	2987
105 Heating steam at FWH5	11.08	361.1	53.54	3179.9
106 Heating steam at FWH6	22.05	217.4	75.74	2682.4
107 Heating steam at FWH7	42.17	254.5	127.38	2803
108 Heating steam at FWH8	67.11	309.6	157.45	2890.7
111 Drain liquid at FWH1	0.12	49.5	200.94	207.3
112 Drain liquid at FWH2	0.58	52.9	180.99	221.5
113 Drain liquid at FWH3	2.4	87.8	121.93	367.7
114 Drain liquid at FWH4	5.18	112	80.99	470.2
115 Drain liquid at FWH5	11.08	184.4	1842.92	782.5
116 Drain liquid at FWH6	22.05	195.6	588.98	832.7
117 Drain liquid at FWH7	42.17	220	513.24	944
118 Drain liquid at FWH8	67.11	256.3	385.86	1116.4
121 Feedwater into FWH1	19.42	34.2	878.88	145
122 Feedwater into FWH2	17.81	47.3	1079.83	199.6
123 Feedwater into FWH3	15.55	82.2	1079.83	345.2
124 Feedwater into FWH4	14.69	106.1	1079.83	446
125 Feedwater into FWH5	11.08	150.5	1079.83	634.5
126 Feedwater into FWH6	253.69	190	1842.92	819.2
127 Feedwater into FWH7	253.13	214.4	1842.92	926.2
128 Feedwater into FWH8	252.53	250.7	1842.92	1090.8
142 Feed water leaving condenser	0.35	33.1	782.74	138.8
143 Cooling water into condenser	3.74	17.7	30275.3	74.5
144 Cooling water leaving condenser	2.51	31	30275.3	130.1
145 Feed water into reactor	252.01	280	1842.92	1230
146 Steam leaving reactor	246.75	499.7	1842.92	3169.6
152 Heating steam of 1st RH	70	313.3	107.95	2893
153 Drain of 1st RH	N/A	N/A	107.95	825.7
154 Heating steam of 2nd RH	246.75	499.7	120.46	3169.6
155 Drain of 2nd RH	N/A	N/A	120.46	1188.2
156 Moisture separator drain	N/A	N/A	120.57	798.4
201 Cooling tower inlet air	N/A	20	32549.72	N/A
204 Cooling tower exit air	N/A	27.2	33201.16	N/A
210 SSR to condenser	1.24	105.8	0.94	2616.5
Valve Stem leak 1 => LPcrs	N/A	N/A	1.13	3167.3
Valve Stem leak 2 => SSR	N/A	N/A	0.05	3167.3
HPT LP leak 1 => FWH4	N/A	N/A	0.67	2583.4
HPT LP leak 2 => SSR	N/A	N/A	0.89	2583.4

The feasibility of the 1,600-MWe turbine-generator was verified with various turbine vendors. BREI has generated a stage-by-stage model of the HPT/IPT and LPT demonstrating that the steam parameters at the turbine inlet (494°C and 23.4 MPa), the steam speeds in the LPT exhaust annulus (<225 m/s), the LPT blade lengths (52") and moisture content (<15%) are all within current standard available technology. Note that the overall size of the SCWR turbine generator is similar to a 1,300-1,500 MWe LWR turbine generator (see Figure 12) because the volumetric steam flow processed in the LPT is similar for both plants.

Candidate materials for all BoP components have been identified by ORNL, and are reported in Table XII. Note that the SCWR builds extensively on the successful materials experience in supercritical fossil plants.

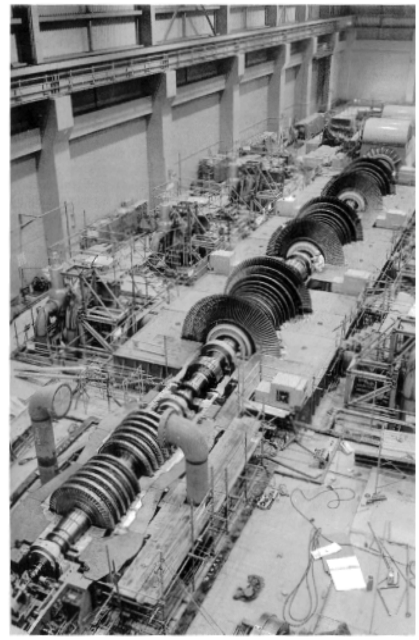


Figure 12. A large LWR turbine generator, from [Logan and Roy 2003]. The SCWR turbine generator is expected to be of similar physical size.

Table XII. Candidate alloys for the SCWR balance of plant.

Component		Fossil	SCWR	Comments
Steam lines		P91 P92	P91 P92	Based on fossil experience New alloy
Turbine	**Casing**	cast 0.5%CrMoV 1.25Cr-0.5Mo 2.25Cr-1Mo P122 (HCM12A)	cast 0.5%CrMoV	Current Developmental
	Valves	cast 0.5CrMoV Cast P91 Cast mod P91+WCoNbB	cast 0.5CrMoV	Current Developmental (EPRI) Developmental (VGB)
	Bolting	1%Cr-Mo-V Type 422: 12%Cr Nimonic 80A	1%Cr-Mo-V 12%Cr	Current Current Current
	Rotor & discs	1%Cr-Mo-V forged NiCrMoV A469 Class 8 NiCrMoV A470 Class 8 NiCrMoV A471 Class 8 Type 422: 12%CrMoV mod 12%CrMoV 9%Cr-Co-Mo-W-V-Nb-N-B	1%Cr-Mo-V	Current, low-alloy, bainitic steels Currently used in Europe Developmental
	Blades	forged Type 403: 12Cr Type 422: 12Cr	Type 403 Type 422	Current
Condenser	**Tubes**	Carbon steel, duplex stainless steels, titanium	Carbon steel, duplex stainless steels, titanium	Based on fossil experience where SCC on coolant side is an issue
	Body	Carbon steel	Carbon steel	Based on fossil experience
Demineralizer/deareator		Carbon steel	Carbon steel	Based on fossil experience
High and low pressure feedwater heaters		Carbon steel	Carbon steel	Based on fossil experience
Condensate and feedwater pumps		F304L	F304L	Based on fossil experience

2.9 Control Strategy

BREI has reviewed the general characteristics, standards and regulations for the control system in existing nuclear power plants in view of the SCWR application. The SCWR presents several similarities and differences with the BWR and PWR systems that affect the control strategy. The BWR similarities are associated with the direct cycle with feedwater flow entering directly into the reactor vessel and steam flow going directly to the turbine. A balance between feed and steam is required to maintain the water inventory in the vessel. Also, soluble poisons such as boric acid cannot be used for reactivity control. The PWR similarities are associated with the high operating pressure, the single-phase conditions at the core outlet and a core outlet temperature that is a function of power and coolant flow. Unique aspects of the SCWR that influence the control concept include the elimination of the recirculation pumps, the low water inventory in the RPV, the large change in coolant density across the core, the absence of a coolant level under supercritical conditions.

The major systems to be controlled in the SCWR are the reactor coolant system, the feedwater and condensate system, the steam system, and the turbine generator system. The main variables to be controlled include the reactor power, the core outlet temperature during supercritical pressure operation (e.g., during full power operation), the reactor pressure, the reactor level during subcritical pressure operation (e.g., during start-up) and the feedwater flow.

Assuming that the SCWR will be operated in a base load rather than a load follow manner, and considering the general characteristics of the SCWR plant, BREI has made the following two recommendations for the SCWR control system:

- *Control of the main reactor variables.* The control rods accomplish the primary control of the thermal power. The turbine control valve provides the control of the pressure, and the feedwater flow (i.e., the feedwater pumps) provides the primary control of the outlet temperature. The control of the coolant inventory in the vessel is accomplished by assuring that steam and feed flow are balanced while maintaining the correct core outlet temperature. The control logic for the main four variables is illustrated in Figures 13 through 16. A detailed discussion of the control strategy can be found in the report prepared by Burns and Roe [BREI 2003].

- *Integrated control system approach.* Rather than an approach in which higher functions such as power or turbine valve control are in manual with lower level control loops in automatic, a coordinated control in which all functions are in automatic is proposed. The relatively small vessel water inventory, the nuclear/thermal-hydraulic coupling, the lack of level indication under supercritical conditions and the absence of recirculation flow makes control more challenging. Thus, the use of an integrated control would allow the system to anticipate changes and react accordingly.

An example of how the SCWR integrated control system will function is provided next. Assume that the reactor is operating at 100% power and the operator decides to lower the power to 95%; then the following sequence will follow:

i. The operator dials in the new thermal power set point.
ii. A signal is sent by the control system to the various control subsystems such as reactor power control, feedwater control and pressure control.
iii. The reactor power control system inserts the control rods.
iv. The core thermal power decreases and the core outlet temperature decreases.

v. The feedwater pumps will reduce the feedwater flow to re-establish the nominal core outlet temperature.
vi. The steam flow will also decrease and thus will the reactor pressure.
vii. The turbine control valve will close slightly to re-establish the nominal pressure.
viii. The system will stabilize at 95% power with a reduced feedwater flow and steam flow and with the nominal pressure and core outlet temperature set points re-established.

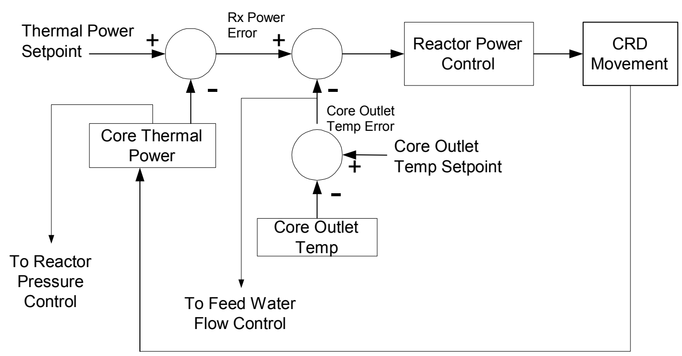


Figure 13. SCWR power control logic.

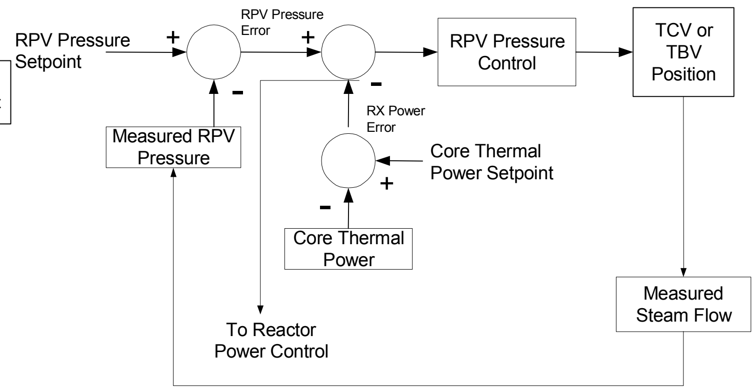


Figure 14. SCWR pressure control logic

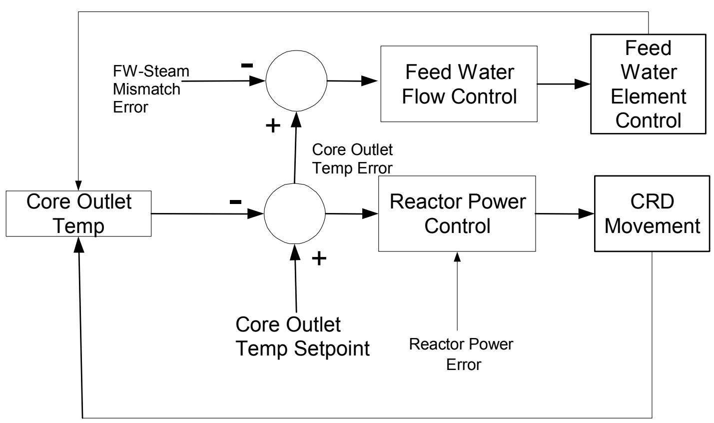


Figure 15. SCWR core outlet temperature control logic.

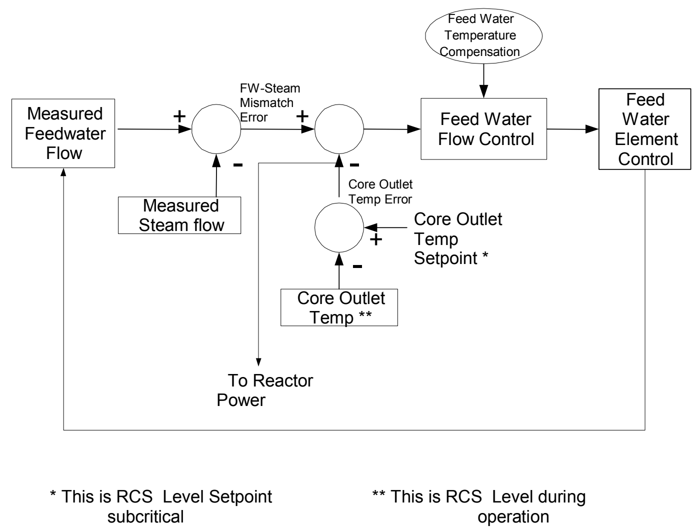


Figure 16. SCWR inventory control logic.

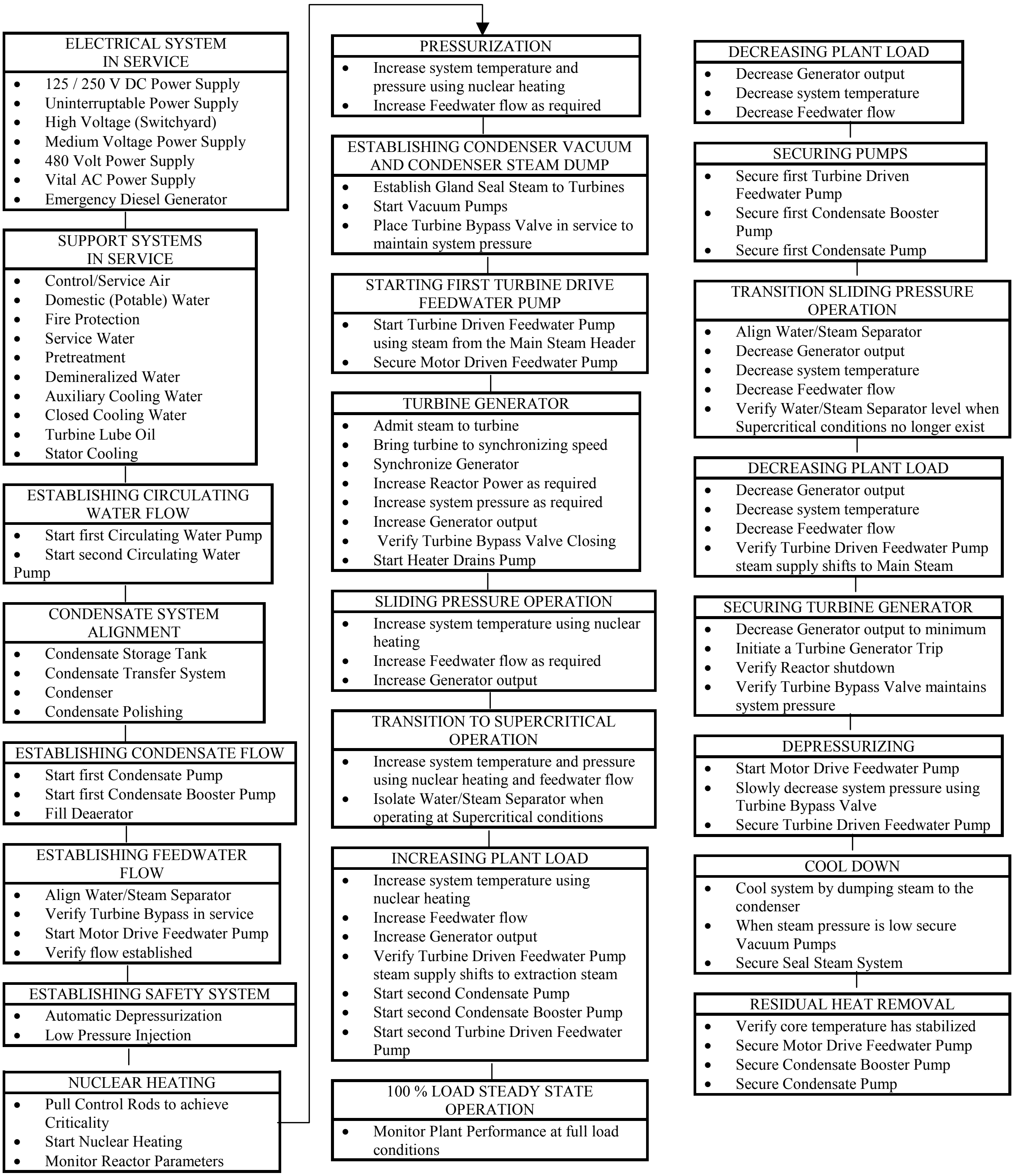
2.10 Start-up Procedures and Related Equipment

There are two general means of control used for once-through supercritical fossil plants. These are constant pressure and variable pressure operation (VPO). Through the 1970's most of the once-through supercritical units were controlled using constant pressure in the boilers. This was due to the fact that these units were predominately base-loaded and U.S. turbines achieved high efficiencies at reduced load by use of high-pressure control stages for partial arc admission (partial steam flow). Therefore the U.S. power generators had little interest in VPO. On the other hand, in Europe and Japan the use of VPO was quite common. Also, U.S. plants were being asked to cycle more frequently and during 1970's

European turbine suppliers started to sell their turbines to U.S. utilities. With this European equipment the U.S. utilities became familiar with VPO and many adopted its practice for their once-through plants. Many plants built in the late 1970's and 1980's included provisions for VPO.

With constant pressure start-up the required components are a start-up turbine bypass, a flush tank and pressure reducing valves. With VPO start-up the required components are a start-up turbine bypass, a steam water separator, drain valves and recirculation pumps. In fossil plants there are three major economic benefits of VPO, i.e., (i) ramping pressure with load reduces turbine life expenditure per cycle, (ii) when turbine temperature change limits unit ramp rate, VPO can allow more rapid load changes than can constant pressure, and (iii) VPO can significantly improve part load heat rate (i.e., thermal efficiency). However, since the SCWR will be a base-load plant these advantages from VPO are somewhat diminished, particularly the low-load heat rate improvements.

As discussed in Section 2.8, there are several variables to be controlled during the start-up and shutdown of the once-through SCWR. The SCWR will require a start-up system to ensure adequate reactor power control over a range of 0.1% to 100% power, adequate control of the reactor water inventory and maintenance of the core outlet temperature via reactor power and feedwater control systems. The system must allow for the warming of steam piping to the turbine and ensure that only dry steam is supplied to the turbine. Given these requirements the use of a hybrid VPO is attractive. The integrated control system being recommended for the SCWR will vary the pressure during start-up, but will do so in discrete steps with about three pressure set-points that will be established by the operator until supercritical-pressure operation is reached. This type of system can be classified as a VPO or sliding pressure approach. A flowchart of the needed sequence and procedures for both start-up and shutdown of the SCWR plant is provided in Figure 17. The procedures for the SCWR start-up will be similar to a LWR except for the transition to and from supercritical-pressure operation. A quantitative simulation of the SCWR start-up procedures including nuclear effects and sizing of the start-up equipment will performed by BREI in FY-04.



(a) Start-up (b) Shutdown

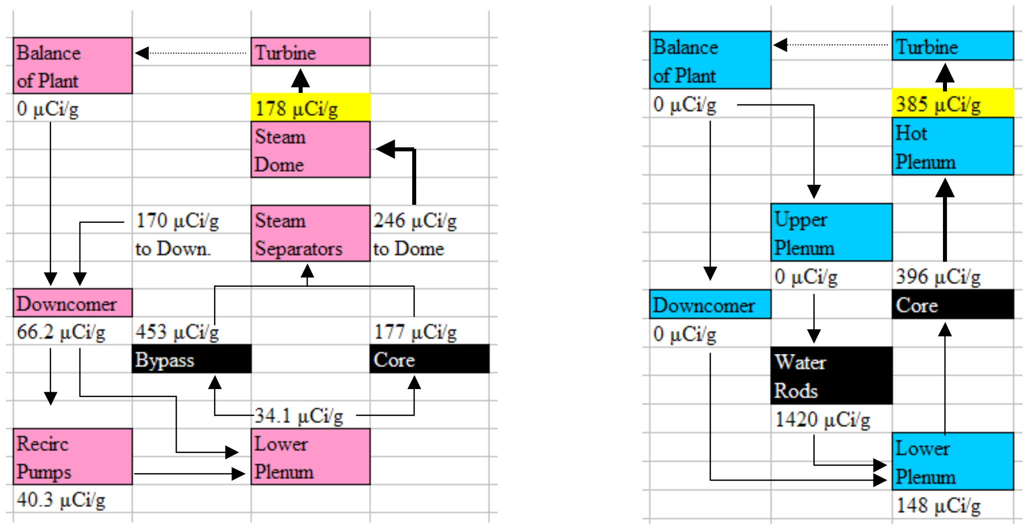
Figure 17. Start-up and shutdown procedures for the SCWR.

2.11 Coolant Activation (N-16)

Because it is a water-cooled nuclear system with a direct thermal cycle, the SCWR shares with the BWR the issue of coolant activation and transport of the coolant activation products to the turbine and balance of plant. Consistent with the BWR experience, the dominant nuclide contributing to the SCWR coolant radioactivity at full power is ^{16}N, which is produced by an (n,p) reaction on ^{16}O [4]. The production and decay of ^{16}N in the SCWR coolant circuit along with the shielding requirements imposed on the balance of plant were analyzed at the INEEL and compared with those in a BWR of similar thermal power rating [Fischer et al., 2003]. The ^{16}N activity distributions in the SCWR and BWR with hydrogen water chemistry are shown in Figure 18. The ^{16}N activities in the steam lines of the SCWR is significantly higher than that of the BWR (~385 vs. ~180 µCi/g) for the following four reasons:

- The coolant transit time in the SCWR core is about twice that of the BWR core,
- The neutron flux is higher in the SCWR because of the higher power density,
- The slow coolant pass in the water rods produces a significant ^{16}N activity at the SCWR core inlet,
- All the ^{16}N generated in the SCWR core is sent to the steam lines because there is no recirculation within the vessel.

A simple gamma attenuation model showed that the higher ^{16}N activity in the SCWR results in shielding requirements only up to 12% higher than for the BWR with hydrogen water chemistry. However, because of the higher SCWR electric power, the specific shielding costs ($/kWe) associated with ^{16}N are expected to be similar to or better than the BWR's.



(a) BWR (with hydrogen water chemistry) (c) SCWR

Figure 18. ^{16}N activity distribution in the BWR and SCWR.

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[4] Note that activated corrosion impurities and some released fission products may also be present in the SCWR coolant, but they were not considered in the study because their contribution to the coolant radioactivity under full-power operation is negligible compared with the radioactivity from coolant activation.

3. Program Management

The Generation-IV SCWR R&D program is managed in the U.S. by the SCWR Product Manager, whose activities include:

1) Generation and maintenance of a reference U.S. SCWR plant design. The design was discussed in Section 2 above and is summarized in a master table, which is continuously updated and periodically distributed to all members of the U.S. team. This approach ensures that the research activities across different functional areas are directed toward the development of a coherent design.

2) Relationship with DOE-NE, which includes periodic reporting on technical progress and costs for the SCWR activities in the U.S., as well as preparation of research plans for future SCWR activities.

3) Representation of the U.S. SCWR team in GIF, which includes membership in the GIF SCWR steering committee and preparation of a GIF R&D plan to coordinate SCWR activities worldwide. In FY-03 the steering committee met four times, in Tokyo (Apr 2003), San Diego (Jun 2003), Mito (Jul 2003) and Kyoto (Sep 2003).

4) Promotion of periodic information exchange meetings to disseminate technical findings. In FY-03 three such meetings have been organized with a broad international attendance in Washington D.C. (Nov 2002), Madison Wisconsin (Apr 2003), and Tokyo Japan (Sep 2003).

5) Promotion of the SCWR concept through the organization of technical sessions at major nuclear engineering conferences. Also, in FY-03 the Product Manager held seminars on the U.S. SCWR activities at the following organizations: General Electric, Westinghouse, Texas A&M, MIT, Idaho State University, University of Wisconsin at Madison, the Ministry of Energy Technology and Industry (METI) of Japan, the conference of Japan nuclear utilities and vendors.

4. Other SCWR Activities in the U.S.

Other non-Generation-IV SCWR activities in the U.S. include four NERI projects and two I-NERI projects. While these projects have well-defined scope and budgets that pre-date the inception of the Generation-IV program, an attempt has been made to coordinate their activities with the Generation-IV activities, so that DOE-NE's objective of assessing the feasibility of the SCWR concept can be pursued effectively. The NERI and I-NERI projects are briefly presented next; however much more information can be found in the progress reports that these projects produce quarterly and annually for DOE.

The first NERI project (01-001) is titled "Feasibility Study of Supercritical Light Water Cooled Reactors for Electric Power Production", is led by the INEEL, started in 2001 and includes Westinghouse, the University of Michigan, and MIT. Activities at INEEL and Westinghouse include the neutronic, thermal-hydraulic, and mechanical design of the SCWR fuel assembly, core and vessel internals. The reference design for this project is the one described in Section 2, but two other approaches eliminating the need for water rods have been explored, one based on the use of solid moderators (zirconium hydrides) and one based on small hexagonal fuel assemblies in which moderation is provided by the feedwater in the inter-assembly gap. As part of this project a new SCW loop was constructed at the University of Michigan to investigate the susceptibility of candidate structural alloys to stress corrosion cracking in supercritical water at various temperature, pressure, oxygen, pH and conductivity conditions. A schematic of the Michigan loop is illustrated in Figure 19. Also an existing SCW loop is being used at MIT for general corrosion screening of candidate alloys (see Figure 20).

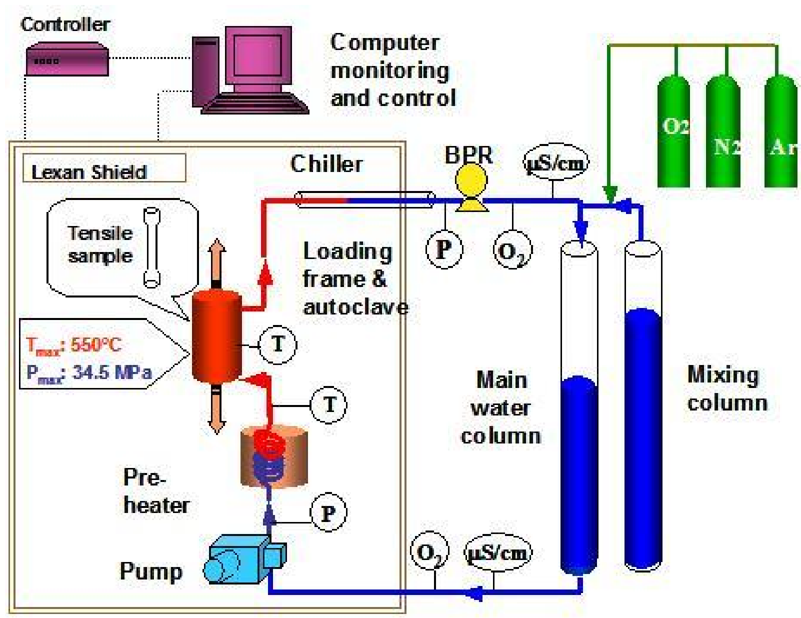

Figure 19. The SCW SCC loop at U-Michigan.

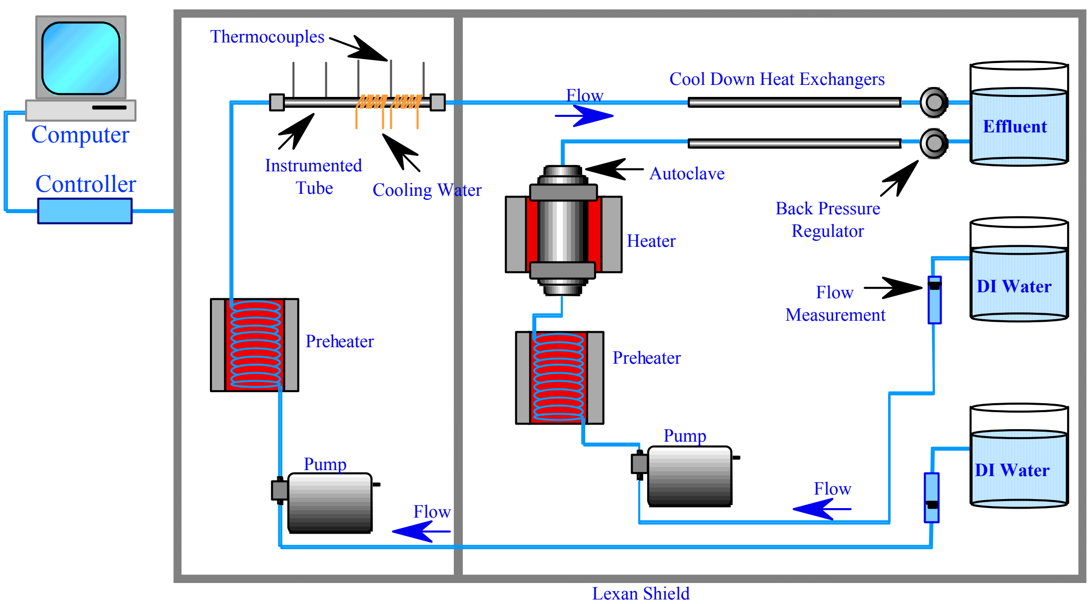

Figure 20. The SCW general corrosion loop at MIT.

A second NERI project (01-091) is titled "Supercritical Water Nuclear Steam Supply System: Innovations in Materials, Neutronics and Thermal-Hydraulics", is led by the University of Wisconsin at Madison and ANL, and also started in 2001. R&D activities include core and plant design with the emphasis on an innovative dual-spectrum core concept using a fast central region surrounded by a thermal annular region. Also, instability experiments are performed in a supercritical CO_2 natural circulation loop (see Figure 21) and corrosion of surface-treated alloys is investigated in a SCW loop (see Figure 22).

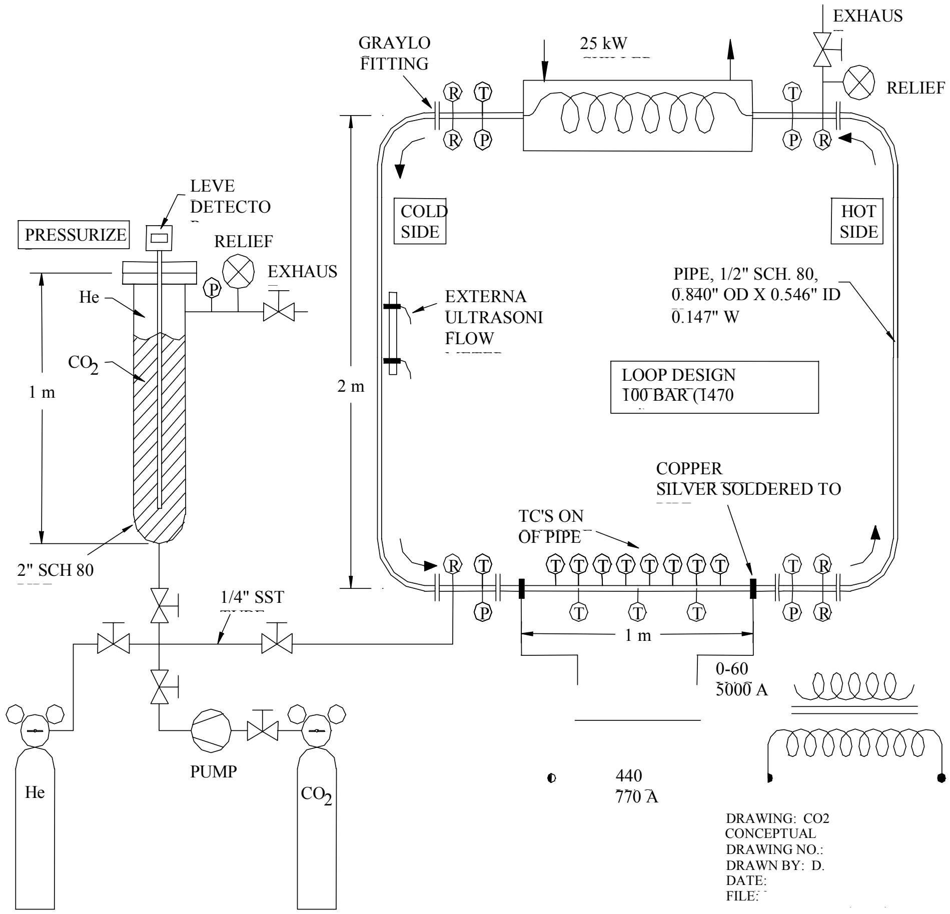

Figure 21. The supercritical CO_2 loop at ANL.

A third NERI project (02-060) is titled "Neutron and Beta/Gamma Radiolysis of Supercritical Water", and is performed at ANL, the University of Wisconsin and Notre Dame University. This project started in 1999 and was renewed in 2002. During the first phase of the project ANL used an accelerator-based pulse radiolysis approach to measure the yields and recombination rates of key radiolytic species in supercritical water. In the second phase the team has built a SCW loop inside the TRIGA reactor at the University of Wisconsin (see Figure 23) to directly measure the concentration of radiolytic species in supercritical water under neutron irradiation. Also, means to suppress radiolysis such as hydrogen injection are being investigated.

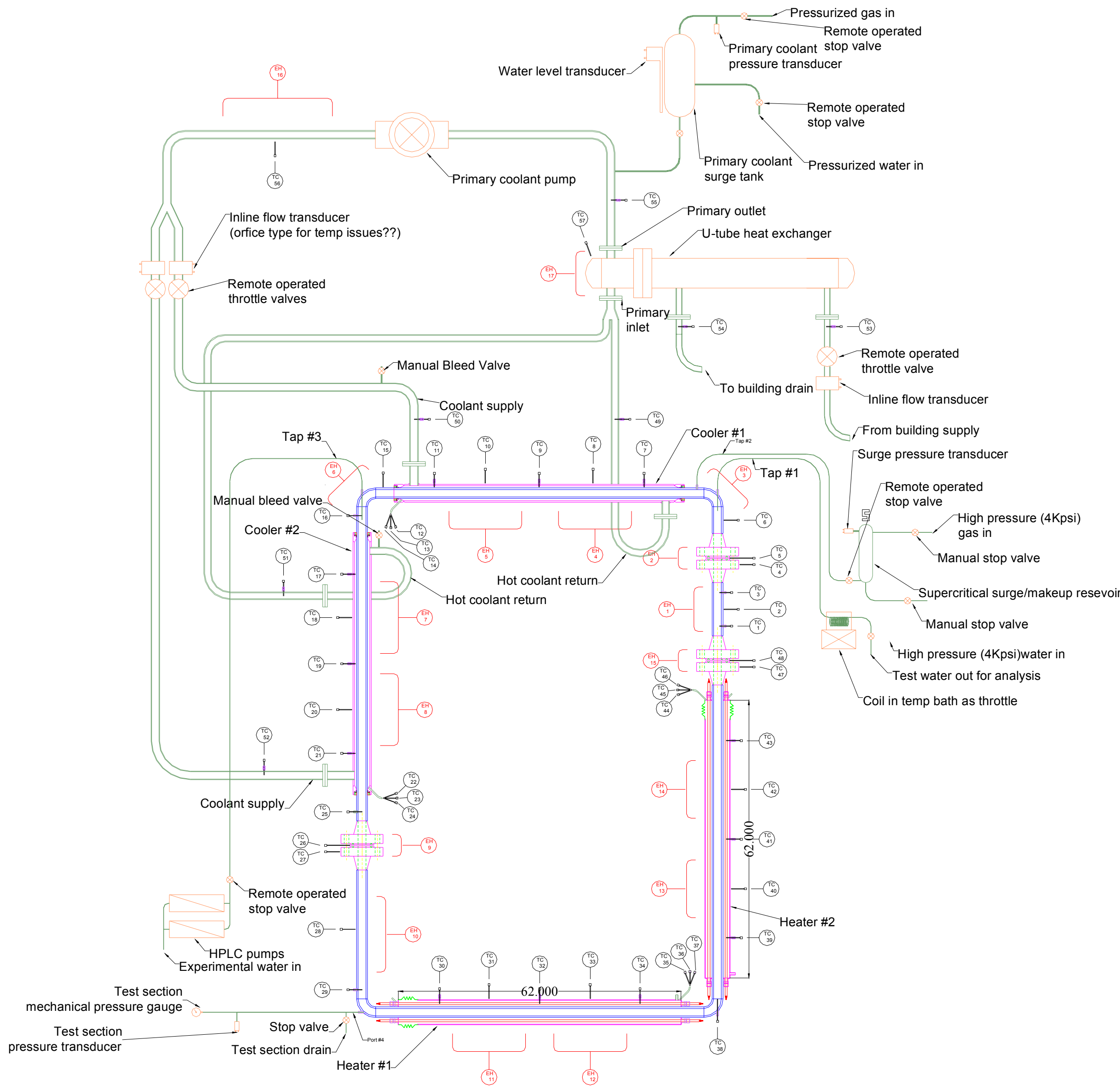

Figure 22. The SCW corrosion loop at U-Wisconsin.

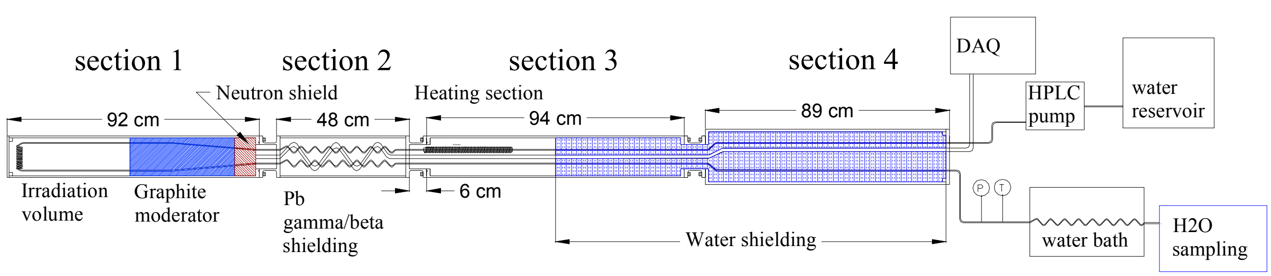

Figure 23. The in-pile SCW loop for radiolysis studies at U-Wisconsin.

The fourth NERI project (01-130) is titled "Fundamental Understanding of Crack Growth in Structural Components of Generation IV Supercritical Light Water Reactors" and is conducted at the Stanford Research Institute International. A test system for electrochemical and fracture mechanics studies in supercritical water was built as part of this project. Controlled-distance electrochemistry is used to measure the transport of ions or ionic defects in the oxide films on structural components made of stainless steels and nickel base alloys at supercritical temperatures. Ionic transport is then correlated with the susceptibility to cracking using fracture surface topography analysis of crack initiation and growth.

The first I-NERI (2003-008-K) is a collaboration with the Korean Atomic Energy Research Institute (KAERI) in South Korea, is titled "Developing and Evaluating Candidate Materials for generation-IV Supercritical Water Reactors", started in 2003 and includes the ANL-West as the lead organization, the INEEL, the University of Michigan and the University of Wisconsin at Madison. The scope of this project includes a survey of commercial alloys for application to the SCWR, preparation of surface-treated coupons at the University of Wisconsin, testing of commercial and surface-treated alloys in the SCW loop at the University of Michigan, as well as mechanical testing of advanced alloys at INEEL.

The second I-NERI (2002-016-K) is also a collaboration with South Korea (Seoul National University and KAIST), is titled "Advanced Computational Thermal Fluid Physics (CTFP) and its Assessment for Light Water Reactors and Supercritical Reactors" and involves INEEL as the lead organization, Iowa State University, the University of Maryland, and Pennsylvania State University. This basic thermal fluids research applies first principles approaches (direct numerical simulation and large eddy simulation) coupled with experimentation (heat transfer and fluid mechanics measurements) to develop reliable computational tools for modeling of transport phenomena in supercritical fluids in complex geometries such as the core of a SCWR. The experimental work is performed at the INEEL's Matched-Index-of-Refraction flow system while the development of the computational methods is performed at the universities.

There exists general consensus that heat transfer data at prototypical SCWR flow and geometry conditions are urgently needed for reliable design of the SCWR core and safety systems. In FY-03 the INEEL performed the conceptual design and detailed cost estimate for a SCW loop that would allow collection of such data [Buongiorno et al. 2003b]. The INEEL loop is shown in Figures 24 and 25 and comprises a heater-bundle test section, preheaters, cooler, pump, accumulator and various auxiliary

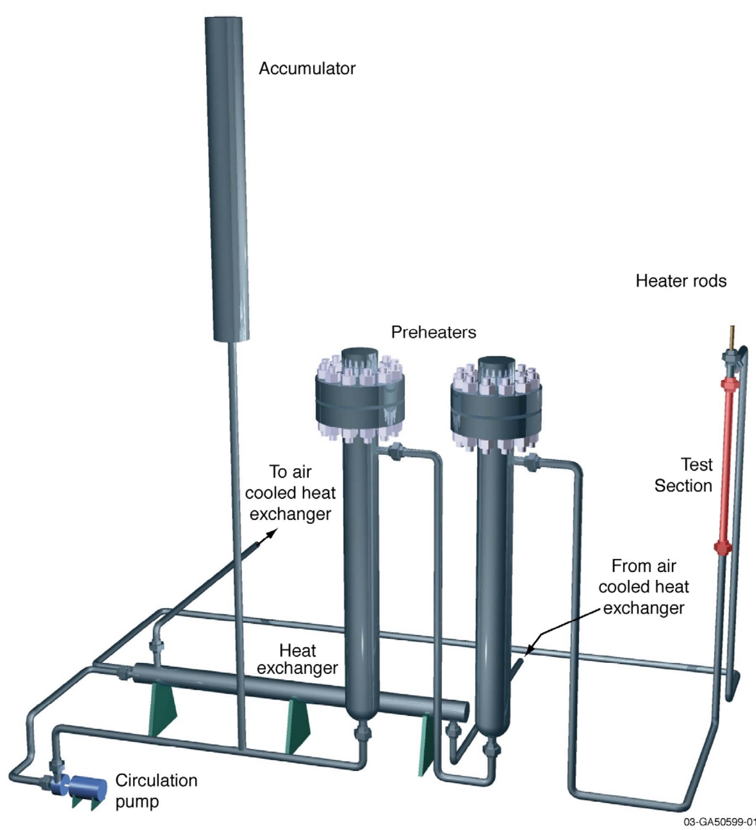

Figure 24. Three-dimensional isometric view of the INEEL SCW heat transfer loop.

systems including a secondary cooling system, a deareator, a water supply system and water discharge tank.

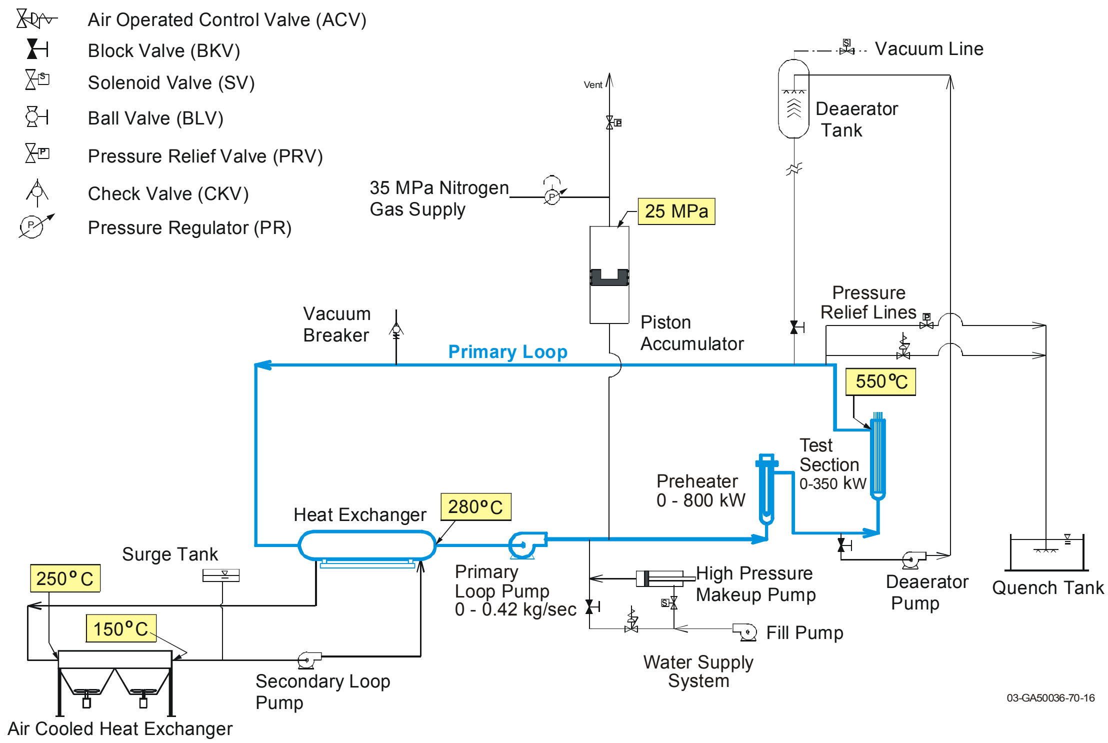

Figure 25. Simplified P&ID of the INEEL SCW heat transfer loop.

5. Conclusions

The research work during the first year of the Generation-IV SCWR program has established that:

1. The SCWR can make substantial use of existing LWR technology in the nuclear island. For example, the design and materials of the SCWR reactor pressure vessel and containment are similar to the PWR's and BWR's, respectively.

2. The SCWR can achieve high thermal efficiencies making extensive use of available supercritical fossil plant technology in the balance of plant. For example, the materials and design of the power conversion cycle as well as the start-up and shutdown procedures and equipment can be drawn from fossil plants with only minor modifications.

3. Based on preliminary one-dimensional analyses, the SCWR appears to be stable with respect to thermal-hydraulic and thermal/nuclear oscillations because of its relatively low coolant reactivity feedback coefficient.

4. The importance of the loss of feedwater as a key abnormal event has been recognized. The design of a suitable high-pressure high-capacity fast-acting auxiliary feedwater system will be a major challenge in proving the viability of the SCWR.

5. Limited corrosion and stress-corrosion testing of traditional stainless steels in high-temperature water has shown that finding materials that would perform satisfactorily in the SCWR

environment will be a challenge. However, classes of materials with promising mechanical properties and corrosion resistance have been identified and will be tested.

In summary, the basic assumptions contained in the Generation-IV Roadmap Report have been confirmed and no new potential showstoppers have been found. The key feasibility issues for the SCWR remain the development of in-core materials and the demonstration of an adequate safety level.

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