SUPERCritical WATER COOLED REACTOR
1 SUPERCRITICAL WATER COOLED REACTOR

1.1 CONCEPT DEVELOPMENT

The Generation IV Roadmap selected the Supercritical Water Cooled Reactor (SCWR) concept as one of the six technologies for further development under Generation IV. GIF has established a Steering Committee for the development of SCWR, with participation of Euratom, Japan, the Republic of Korea and Canada. Nevertheless there is an active and large development program in the United States of America to develop a supercritical nuclear power system.

Buongiorno and MacDonald in their 2003 report, describe in a very exhaustive way the SCWR reactor concept and its perspectives with these words: “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.”

On the European side, several European Organizations are collaborating together in the framework of the High Performance Light Water Reactor (HPLWR) project which aims to demonstrate the feasibility to design a competitive and safe supercritical power reactor which complies with all GEN IV goals and gives assurance of investment protection. These institutions are:

- Forschungszentrum Karlsruhe (FZK), Germany
- Commissariat à l’Energie Atomique (CEA), France
- AREVA NP GmbH, Germany
- University of Stuttgart, Germany
- KFKI Atomic Energy Research Institute, Hungary
- KTH Stockholm, Sweden
- Nuclear Research and Consultancy Group (NRG), The Netherlands
- Paul Scherrer Institute (PSI), Switzerland
One of the principal purposes of the GIF R&D plan is to identify the priorities for common and coordinated SCWR research. The major topics are the system design, fuel development, supercritical coolant technology and materials, component development, balance of plant, the hydrogen production and the demonstration. The matrix of the research activities regarding different design options that are carried out by the international research community is shown in Figure 1.1. Both Thermal and Fast neutron spectra reactor systems are under investigation so that the possible fuel cycle will be once trough or a closed one. In particular Canada is developing activities on thermal neutron/pressure tube reactor aiming to the application of the know-how gained in designing and operating the CANDU reactor to a future CANDU SCWR reactor. The different systems and their characteristics will be explained in the next section.

![Figure 1.1: SCWR design options](image)
1.2 TECHNICAL ASPECTS

Different concepts for the design of a SCWR are under study and they are divided essentially in two different groups:

- pressure tube system
- pressure vessel system

The first reactor type is derived by the CANDU experience and for its development the main relevant features of the design are retained.

The second reactor type derives from the experience gained designing and operating the existing light water cooled reactors.

1.2.1 CANDU SCWR

The CANDU SCWR (Figure 1.2) is a 1220 MWe (2540 MWth) supercritical reactor, with a previewed efficiency of the 48%. Although SCWR can be designed as fast or thermal reactors, the CANDU SCWR is a thermal reactor.

![Figure 1.2: CANDU SCWR Reactor and its “products”](image)

The working pressure of the coolant inside the tube immersed in the moderator is 25 MPa. The coolant inlet temperature is 350 °C and at the outlet could reach the 650°C higher than that of the other water cooled reactors like shown in Figure 1.3.
The reactor is moderated using heavy water so that it will be possible to use natural uranium $\text{UO}_2$ as fuel without enrichment as much as Thorium. The coolant fluid is light water. In particular, a peak cladding inner surface temperature of $<850 \, ^\circ\text{C}$ has been set as an objective of the preliminary design.

Conditions and other parameters for CANDU SCWR are included in Table 1.1.

Table 1.1: CANDU SCWR preliminary specifications

<table>
<thead>
<tr>
<th>Spectrum</th>
<th>Thermal</th>
</tr>
</thead>
<tbody>
<tr>
<td>Moderator</td>
<td>Heavy water</td>
</tr>
<tr>
<td>Coolant</td>
<td>Light water</td>
</tr>
<tr>
<td>Thermal Power</td>
<td>2540 MW</td>
</tr>
<tr>
<td>Flow Rate</td>
<td>1320 kg/s</td>
</tr>
<tr>
<td>Number of Channels</td>
<td>300</td>
</tr>
<tr>
<td>Electric Power</td>
<td>1220 MW</td>
</tr>
<tr>
<td>Efficiency</td>
<td>48%</td>
</tr>
<tr>
<td>Fuel</td>
<td>$\text{UO}_2$/Th</td>
</tr>
<tr>
<td>Enrichment</td>
<td>4%</td>
</tr>
<tr>
<td>Inlet Temperature</td>
<td>$350,^\circ\text{C}$</td>
</tr>
<tr>
<td>Outlet Temperature</td>
<td>$625,^\circ\text{C}$</td>
</tr>
<tr>
<td>Cladding Temperature</td>
<td>$&lt;850,^\circ\text{C}$</td>
</tr>
<tr>
<td>Calandria Diameter</td>
<td>4 m</td>
</tr>
</tbody>
</table>

The thermophysical properties of water for the CANDU-SCWR, the current CANDU-6 and the Pressurized Water Reactor (PWR) are given in Table 1.2.
Table 1.2: Comparison of the Values of Thermophysical Properties of Water and Values of Heat Transfer Coefficient for the Conditions of CANDU-SCWR, CANDU-6 and a generic PWR

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Unit</th>
<th>CANDU-SCWR</th>
<th>CANDU-6</th>
<th>PWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure</td>
<td>MPa</td>
<td>25**</td>
<td>10.5</td>
<td>15</td>
</tr>
<tr>
<td>Temperature</td>
<td>°C</td>
<td>Inlet 350</td>
<td>Outlet 625</td>
<td>Inlet 265</td>
</tr>
<tr>
<td>ΔT from inlet to outlet</td>
<td>°C</td>
<td>275</td>
<td>45</td>
<td>85</td>
</tr>
<tr>
<td>Density</td>
<td>kg/m³</td>
<td>625.3</td>
<td>67.58</td>
<td>782.9</td>
</tr>
<tr>
<td>Enthalpy</td>
<td>kJ/kg</td>
<td>1624</td>
<td>3567</td>
<td>1159</td>
</tr>
<tr>
<td>Increase in enthalpy</td>
<td>kJ/kg</td>
<td>1943</td>
<td>242</td>
<td>501</td>
</tr>
<tr>
<td>from inlet to outlet</td>
<td>kJ/kg K</td>
<td>7.06</td>
<td>5.38</td>
<td>5.74</td>
</tr>
<tr>
<td>Specific heat</td>
<td>J/kg K</td>
<td>6978</td>
<td>2880</td>
<td>4956</td>
</tr>
<tr>
<td>Expansivity</td>
<td>1/K</td>
<td>5.17·10⁻⁴</td>
<td>1.74·10⁻⁴</td>
<td>2.09·10⁻⁴</td>
</tr>
<tr>
<td>Thermal conductivity</td>
<td>W/m K</td>
<td>0.481</td>
<td>0.107</td>
<td>0.611</td>
</tr>
<tr>
<td>Dynamic viscosity</td>
<td>Pa·s</td>
<td>7.28·1⁵</td>
<td>3.55·1⁵</td>
<td>10.12·1⁵</td>
</tr>
<tr>
<td>Kinematic viscosity</td>
<td>m²/s</td>
<td>11.63·1⁴</td>
<td>52.47·1⁴</td>
<td>12.92·1⁴</td>
</tr>
<tr>
<td>Diffusivity</td>
<td>m²/s</td>
<td>11.02·1⁴</td>
<td>54.72·1⁴</td>
<td>15.75·1⁴</td>
</tr>
<tr>
<td>Surface tension</td>
<td>N/m</td>
<td>–</td>
<td>–</td>
<td>22.5·1⁴</td>
</tr>
<tr>
<td>Prandtl number</td>
<td>–</td>
<td>1.06</td>
<td>0.96</td>
<td>0.82</td>
</tr>
<tr>
<td>Reynolds number (× 10⁶) at</td>
<td>G=860 kg·m⁻³, Dp=8 mm</td>
<td>0.946</td>
<td>1.940</td>
<td>0.680</td>
</tr>
<tr>
<td>Nuettel number*** (0.023 Re²/Pr²)</td>
<td>–</td>
<td>1148</td>
<td>2425</td>
<td>985</td>
</tr>
<tr>
<td>Heat transfer coefficient</td>
<td>W/m² K</td>
<td>8527</td>
<td>3228</td>
<td>7522</td>
</tr>
</tbody>
</table>

¹ All thermophysical properties of water were calculated according to NIST (2002).
² Critical point for water is 374°C and 22.12 MPa.
³ This value of mass flux corresponds to CANDU-SCWR operating conditions. Mass flux values in subcritical nuclear reactors are much higher, therefore, values of Reynolds number, Nusselt number and heat transfer coefficient will be also much higher in subcritical reactors.
⁴ Nusselt number is calculated using the Dittus-Boelter correlation (1930) for forced convective heat transfer in a circular tube as a first estimate only.

A sketch of the working principle of a pressure tubes reactor is shown in Figure 1.4. Obviously in case of supercritical condition the steam generator is not present.

Figure 1.4: Working principle of a pressure tubes reactor
According to Chow and Khartabil of the AECL (Atomic Energy of Canada Limited) the advantages of a CANDU SCWR reactor could be:

- Modular, replaceable horizontal fuel channel
- Simple, economical fuel bundle design
- Flexible fuel cycle with high neutron efficiency
- Separate cool, low pressure heavy water moderator with back-up heat sink capability
- No large average coolant density difference in a normal section of the core

Several fuel bundle design have been investigated for the application and Chow and Khartabil in their work report information for two different designs:

- the High Efficiency Channel (HEC)
- the Re-Entrant Fuel Channel

The HEC (Figure 1.5) does not use a calandria tube to separate the pressure tube from the moderator. Each pressure tube is in direct contact with the moderator, which operates at an average temperature of about 80°C. The pressure tube is thermally insulated from the hot coolant by an insulator. A perforated metal liner protects the insulator from being damaged by the fuel bundles and from erosion by the coolant flow. The coolant pressure is transmitted through the perforated metal liner and small openings in the insulator and directly applied to the pressure tube. The insulator does not need to support the coolant pressure, although it must be able to withstand the weight of the fuel bundles.

Figure 1.5: Section of the High Efficiency Channel (HEC)
The Re-Entrant Fuel Channel (Figure 1.6) requires instead a calandria tube to separate the pressure tube and the moderator. The fuel channel consists of a pressure tube and a concentric inner tube. Similar to the current CANDU reactors, the pressure tube of this design is separated from the heavy water moderator by a gas annulus. SCW coolant flows first between a pressure tube and the inner tube. It turns around and flows through the inner tube, where the fuel resides. This design keeps the pressure tube at a temperature of about 350°C to 400°C.

Figure 1.6: Scheme of the Re-Entrant Fuel Channel

1.2.2 Pressure Vessel type SCWR: HPLWR

In order to present this reactor concept type, we will concentrate our attention on the European High Pressure Light Water Reactor (HPLWR). This concept is based on a ten years experience at the University of Tokyo, and it is under study since 2000 by an international team of research centers and industries to assess its feasibility and the future potentials.

The HPLWR presents an once-through supercritical steam cycle with 25 MPa design pressure and >500°C turbine inlet temperature. The overall specification of this system (Figure 1.7) are:

- Net electric power 1000 MW
- Thermal Power 2280 MW
- System Pressure 25 MPa
- Core inlet temperature 280°C
- Core exit temperature > 500°C
- Core height 4.2 m
- Safety systems active and passive
The main advantages of the HPLWR individuated by Starflinger (FZK) in its presentation on the European activities aiming at the design of a supercritical reactor at the ICAPP07 are:

- Single phase fluid, therefore no risk of a boiling crisis
- High steam enthalpies, causing high cycle efficiencies and high specific turbine power
- Once through steam cycle, avoiding steam generators and primary or recirculation pumps
- Use of proven turbine technologies from fossil fired power plant
- Evolution from existing water cooled reactors
- Use of existing UO₂ or MOX fuel
- Improved economics, better fuel utilization
- Level of safety as high or even higher than 3rd Generation

Problems concerning the design, the technologies and the materials to be adopted will be discussed in the next section.
1.3 TECHNICAL PROBLEMS

Materials issues possess a particular salience within the SCWR context. No nuclear reactor has yet been built using supercritical water as the coolant, while for the other Generation-IV concepts, even the VHTR, demonstration or experimental reactors of very closely related concepts have already been built, and coolant-material interactions are less uncertain.

In order to make the collaborative R&D research more productive Schulenberg reports a list of characteristics that should be considered in developing the different design concepts:

- The coolant is light water with a core outlet temperature <625°C to take full advantage of available fossil fired power plant technology in the balance of plant. This is consistent for the designs with core outlet temperatures around 500°C, currently under investigation in most GIF countries.
- A once-through, direct cycle will be utilized to eliminate the need of steam generators, steam separators and dryers, pressurizers, and recirculation pumps.
- The core will be based on a thermal neutron spectrum. This will limit the damage in structural materials to 10⁻³₀ dpa (displacement per atom).
- A once-through fuel cycle based on traditional UO₂ or mixed UO₂–PuO₂ fuel pellets will be used.
- Materials and chemistry used in the fossil fired power plant industry are the starting point for future testing.
- SCWR is a base load electricity producer.

Baindur in a paper published in March 2008 for the Bulletin of the Canadian Nuclear Society reports that although many fossil-fuel-fired power plants using supercritical water have been built, there is no precedent for the type of radiological-thermochemical stress that materials in the SCWR reactor core will face, especially in a fast spectrum fuel cycle. Though materials eventually to be used for the SCWR are expected to evolve from those currently being used in nuclear reactors generally, the issues specific to SCWR do warrant a detailed investigation.

Also Chow in his presentation for the SCWR-2007 conference in Shanghai, pointed out that data on irradiated material for the new fuel channel materials have not yet
been produced and in his paper reports that for the new materials in-flux studies with appropriate coolant chemistry will be required, as neutron irradiation can have significant effects on the behavior in a supercritical coolant. He also points out that no alloy or material has received enough study to ensure its viability in an SCWR. Extensive R&D on materials will need to be conducted on candidate materials in the following areas:

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

Canada concentrates on a pressure tube design like the well known CANDU reactor, in which the high pressure tubes must be thermally insulated from inside to avoid high pressure and temperature loading of the tubes and to minimize heat up of the surrounding heavy water moderator. Schulenberg et al. in the FISA2006 paper, report that this concept offers highest flexibility with respect to the flow path through the core. Furthermore they report that other research institutions prefer to study reactors of the pressure vessel type. In this case the large density drop of the coolant requires additional moderator, like for example solid zirconium hydride (this is the case of the Korean concept).

Heat transfer of supercritical water in tubes is well known today from fossil fired boiler development. One of the most important experimental apparatus in this field is located in Erlangen (Germany). It is the BENSON test rig (Figure 1.8).

![Figure 1.8: BENSON test rig in Erlangen](image-url)
The test matrix (Figure 1.9) gives an overview of the measurements which have been systematically performed since 1975 for a wide range of parameters. These measurements enabled generation of scientifically founded design documents as well as increasing heat transfer in rifled tubes by approximately 50% through optimization of rifling geometry.

![Test Matrix Diagram]

**Figure 1.9: BENSON Boiler Heat Transfer and Pressure Drop in Tubes experimental database**

In the FY2005 report of the Idaho National Laboratory it is reported that this facility at the Framatome-ANP laboratories was used for supercritical fossil boiler tube tests and could be used by GIF for single-tube experiments. This facility also has a sufficiently large power supply and a pump to accommodate a relatively large heated-rod bundle. However, the actual bundle test section would have to be built as part of SCWR projects. The work will consist of the following elements:

- Upgrade of the Erlangen facility (the U.S. will design and construct the bundle test section).
- Measure the heat transfer coefficient at prototypical SCWR flow and geometry conditions.

In the Schulenber et al. FISA 2006 paper a chart of the planned HPLWR project is shown. Figure 1.10 reports the chart of the project 5 – Heat transfer. As you can see all the activities regarding the design, the construction and the operation of the new test section in Erlangen were planned to finish by the end of June 2008. No information has been collected about the effective stage of the work.
After, calculation and benchmark activities were planned. It has not been possible to collect information on the real state of the work in order to know if experimental data have been acquired or if the work has been for some reason delayed.

![Figure 1.10: Chart of the planned HPLWR project, WP5 Heat transfer](image)

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

In fact traditional thermal-hydraulic codes are not generally suitable for the analysis of SCWR. They need changes, for sure, to improve numerical stability, in order to handle the large variations of the thermo-physical properties like it happens in case of supercritical water. The codes are not yet verified and validated for their application to SCWR analysis. Further, like it is reported in the FY2005 report of the Idaho National Laboratory, the correlations and models implemented by the codes are not suitable to describe key SCWR thermal phenomena such as deterioration of heat transfer or critical flow at supercritical pressure. In addition, the codes are not validated for the tighter and more heterogeneous core geometries typical of SCWR designs with water rods. Furthermore, a transition to two-phase flow conditions in the core has to be taken into account, which will happen in case of depressurization of the reactor pressure vessel, such as during normal shutdown or in emergency cases. The codes have to be capable to model such transitions. For these and other reasons a huge activity in this field has to be done.
1.4 ECONOMIC ASPECTS

A SCWR reactor is clearly intended to be an evolution of current light water reactors using latest technologies of fossil fired power plants. As a baseload electricity producer, its main advantages shall be to lower the specific investment costs and to increase the thermal efficiency compared with latest PWR, BWR or CANDU design.

Essentially all the material regarding economic issues for the SCWR was elaborated by Bittermann at al. in different works. They proposed the same conclusions in different, papers, presentations, official documents. After a detailed analysis based on the methodology presented in the OECD/NEA book titled “Reduction of Capital Costs of Nuclear Power Plants”, Bittermann et al. elaborated in 2003 the conclusions reported in the following paragraph. The formulation and the defining of the problem are reported in Annex 1.

The conclusion that Bittermann et al. reached in their work is that, the estimated cost reductions for the HPLWR compared with the defined reference plant (see Annex1) are: 30% reduction for building and structures, 35% reduction for the reactor plant, 10% reduction for the turbine plant, and 20 to 25% reduction in overnight capital cost. An initial economic target for the HPLWR is set at 1000 €/kWe and 3-4 cent/kWh levelized generation cost.

Such nuclear plants may have the potential to reach electricity generation costs which can significantly increase the economic advantages of nuclear plants compared with fossil fueled plants. This statement especially holds true if one considers in the future a further increase in fossil fuel prices and the possibility of getting deeper insight into the design of HPLWRs in order to explore the potential of cost reduction more precisely and in more detail. Considering the current technical status of the HPLWR, this evaluation is considered to be substantiated enough to justify the continuation of the HPLWR development work.

Once this economic target will be reached it could be possible to think about the use of the off-peak energy that will be produce by the NPP because of its base-load characteristics. Important and interesting under this point of view are the consideration that Fischer did more than twenty years ago regarding possible way of producing hydrogen with a base-load NPP that are reported here under.
It is clear that in the medium term only relatively small surplus capacities from nuclear power plants will be available, exceptions, however, are those countries with an already high share of nuclear capacity in their electricity supply system, e.g. France, Belgium, Sweden and Switzerland. It could be possible in order to give an impulse to the creation and the broadening of the so called and “hydrogen economy”, to introduce this surplus energy into the grid to produce in a distributed way the hydrogen directly or quasi-directly at the refueling stations avoiding all the difficulties of a hydrogen long distances transportation system. Hydrogen could be produce even with electrolysis plants. This solution takes more sense if it is thought in connection to the frequency control of network. The electrolysis of water could be realized in a modulated/discontinued way.

In fact, as Fischer said, nuclear power units used for frequency control of the grid cannot be operated at full power but at about 95% only. The newly developed H₂/O₂ steam generator for the immediate additional steam supply (stationary working conditions of this component are reached within one second) offer that the corresponding power units can be operated again at full power, whereas the frequency control is done by short-time overload operation, carried out instantly by steam from H₂/O₂ - steam generators. The necessary H and O can be produced by electrolysis of water in off-peak periods. In each large grid system there are reserve capacities available, which can be used via interruptible electrolysis plants. In this way the necessary reserves are nevertheless available for the grid as spinning reserves or for peak load management. In summary, of specific interest could be already in the medium term the following:

- Electrolysis of water, based on interruptible bulk power system electricity at incremental cost.
- Balancing of grid loading, reversal of load-following operation
- Application of base load capacity for frequency control at 100% full power by using H₂/O₂ steam generators for immediate additional steam supply
- Surplus and reserve power plant capacities for H₂/O₂ production and utilization as spinning reverse and peak load management systems
1.5 ENVIRONMENTAL AND SOCIAL ASPECTS

Since the project state of SCWR reactor is only at the conceptual state and a nuclear power plant with a similar operating concept has never been built, it is difficult to define a possible response of the public opinion. It is possible to say that the SCWR is not a completely revolutionary idea because it is the application of the very well known technology of the working fluid in supercritical condition at a nuclear power plant.

A SCWR power plant could be proposed like the mix of the best part of the fossil fuel and nuclear technology. In fact the advanced, safe, reliable technology of the light water reactors will meet the clean, high performance, technology of the supercritical steam cycle.

Nowadays Germany owns the primacy in the world on the technology of supercritical boilers. In fact leading boiler manufacturers construct BENSON boilers worldwide under Siemens license. It means that in Germany it could be possible to take advantages of this fact in order to promote further research or to promote the management of a future pilot plant. There is also a very huge amount of experimental data that has been collected through the year by the use of the BENSON test rig situated in Erlangen.

The combined use of these technologies will allow a higher performance (ca. +10%) of the power plant, this will result in a better exploitation of the nuclear fuel in comparison with a normal LWR. Even if the performance of the supercritical power plant will be then comparable to that reachable by the use of the newest fossil fuel technologies, this nuclear application boasts the possibility of producing energy without the production of polluting agents. It is also to consider that the nuclear fuel cycle of an SCWR will be a once-trough, thermal spectrum nuclear reactor (even if the fast neutron spectrum is a still an open option) and that it is rated as good by the GIF in safety and in proliferation resistance and physical protection. The application of this reactor, in the future, will take sense together with other reactors that will allow the closure of the nuclear fuel cycle. In fact, only thinking at the nuclear sector as a whole and not at each plant as a separate part, could be possible to reach high level of sustainability that will allow the nuclear energy to gain the place it deserves among all other energy production options.
The SCWR reactor, concerning the fuel cycle, the wastes and their management does not add anything new to the actual situation of LWRs since issues on spent fuel management and disposal remains.

Given the daring economic objectives declared by the participant of the European HPWLR project, This technology will allow the generation of electric energy continuously at a very low cost so that it could be possible to give an impulse to the production of another very precious energy vector: the hydrogen. It will translate in space and in time the energy content of the electric energy produced by the plant during the low-load period.
1.6 BIBLIOGRAPHY

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1.7 SUPERCRITICAL WATER COOLED REACTOR: ANNEX 1

The formulation used by Bittermann et al. is based on these considerations:

- defining a plant
- using it as scale and identify the essential differences of a HPLWR plant according to the present status of knowledge related to this “scale plant”
- evaluating these differences in terms of relative cost changes.

Since the HPLWR is considered to be closer to a BWR from the design point of view a cost structure for a BWR is used. Like reference for a modern BWR, the ABWR is taken. The results of a costs breakdown comparison between these reactors is shown in Table 1.3.

Table 1.3: Cost breakdown comparison

<table>
<thead>
<tr>
<th>Cost Item</th>
<th>ABWR</th>
<th>PWR Reference</th>
<th>“Scale plant” for HPLWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Direct Costs</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Buildings and structures</td>
<td>20</td>
<td>18</td>
<td>19</td>
</tr>
<tr>
<td>Reactor plant</td>
<td>24</td>
<td>25</td>
<td>24</td>
</tr>
<tr>
<td>Turbine plant</td>
<td>10</td>
<td>14</td>
<td>9</td>
</tr>
<tr>
<td>Electrical plant</td>
<td>7</td>
<td>11</td>
<td>9</td>
</tr>
<tr>
<td>Miscellaneous plant</td>
<td>2</td>
<td>7</td>
<td>9</td>
</tr>
<tr>
<td>Direct Costs Total</td>
<td>63</td>
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<td>Indirect Costs</td>
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</tr>
<tr>
<td>Construction services</td>
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<tr>
<td>Project management</td>
<td>3</td>
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<td>5</td>
</tr>
<tr>
<td>Site management and</td>
<td>8</td>
<td>3</td>
<td>6</td>
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<tr>
<td>commissioning</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Indirect Costs Total</td>
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<td>23</td>
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<tr>
<td>Other Costs</td>
<td>15</td>
<td>11</td>
<td>7</td>
</tr>
</tbody>
</table>

There direct costs, indirect costs and other cost are defined in Table 1.4.
Table 1.4: Definition of direct, indirect, and other costs

| Direct Costs | 1.1 Land and land rights  
1.2 Reactor plant equipment  
1.3 Turbine generator plant equipment  
1.4 Electrical and I&C plant equipment  
1.5 Heat rejection system equipment  
1.6 Miscellaneous plant equipment  
1.7 Construction at the plant site |
| Indirect Costs | 2.1 Design and engineering services  
2.2 Project management services  
2.3 Commissioning |
| Other Costs | 3.1 Training and technology transfer  
3.2 Taxes and insurance  
3.3 Transportation  
3.4 Owner’s costs  
3.5 Ore ore costs  
3.6 Contingencies |
| TOTAL CAPITAL COSTS (means overnight costs since financial costs are not included) | Direct Costs + Indirect Costs + Other Costs |

Then, in order to further calculate the capital costs the following assumption are done:

- Major cost reductions expected in comparison to a reference BWR:
  - no steam separators
  - no steam dryers
  - no main coolant circulation pumps
  - smaller RPV (Reactor Pressure Vessel)
  - smaller containment
  - smaller reactor building
  - reduction in number of steam lines
  - reduction of spent fuel storage pool volume
  - smaller condenser
  - smaller cooling tower
  - reduction of systems capacity due to higher efficiency and consequently reduction in building volume
  - reduction in I&C costs due to simple safety concept including passive features
  - reduction of crane capacity due to smaller weight of components
  - reduction in capacity of HVAC due to smaller building volumes
- Reduction of fire protection measures due to smaller building volumes

- Major cost increase expected in comparison to a reference BWR:
  - Equipment for start-up (flash tank, valves, piping) plus
  - Additional corresponding I&C equipment
  - Design of specific components and systems for higher pressure and temperatures
  - Steam reheater with extended design conditions
  - Complicated fuel assembly

Further, there is a big list of other potential cost reductions that derives from a specific design of the reactor and they are:

- Increase in plant size
- Improvement in construction methods (use of modules)
- Reduction in construction schedule
- Improvement in designs and design and analysis methods
- Improvement in procurement (organization and contractual aspects)
- Standardization and construction in series
- Construction in multiple units
- Regulatory and policy reform (design basis for safety systems, components and structures)

In Table 1.5 are reported the costs reduction that is possible to obtain applying the assumption listed here above, in comparison with “Scale Plant” values.
Considering now the fuel cycle costs, the other parameters that have a considerable effect on the calculation of the costs are:

- thermal efficiency
- average discharge burn-up
- enrichment that is necessary to achieve the burn-up and fuel cycle length
- fuel assembly fabrication costs

Parameters that have been neglected are:

- interest rate
- discount rate
- higher price of fuel assembly fabrication to achieve a
- higher burn-up (at the moment no commercial facility exists which can produce fuel at enrichment higher than 5%)

Some data which are important for the calculation of the fuel cycle costs and which are currently used for calculation of fuel cycle costs for other plants are as follows (2003 price level):

- price of natural uranium 30 US $/kg
- conversion costs 5 US $/kg
- enrichment costs 80 US $/SWU (SWU = Separative Work Unit)
• tails assay 0,3% U235
• disposal costs 500 US $/kg U – 2000 US $/kg U (they depend on the different countries)

For the calculation of the HPLWR fuel cycle costs the following plant specific parameters are selected:
• thermal efficiency 44%
• average discharge burn-up 60 GWd/kg U
• fuel cycle length 440 d
• average enrichment 6,2%
• fuel assembly fabrication cost 250 US $
• disposal costs 500 US $/kg U

Using the above described values for the calculation of a 1000 MWe HPLWR, the fuel cycle cost is 3,08 mills/kWh, that results in a 10% savings of fuel cycle costs compared to LWRs. It should also be considered that higher Uranium price will not affect so much this calculation.

We assume that the reference plant can reach the capital cost target as it is considered for the EPR which is calculated as 1250 €/kWe and the resulting electricity generation costs which are depicted in the following table, then electricity generation costs of about 2.6 cent/kWh could be expected for the HPLWR.