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Core design feature studies and research needs for high performance light water reactors

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Abstract – *Light water reactors using super-critical water as coolant differ from conventional PWR or BWR design by higher plant efficiencies, but also by smaller plant size at comparable electric output. These advantages are partly compensated by new aspects such as the larger coolant temperatures in the core, or the larger coolant density differences which require several new and unconventional design features that need to be addressed in a correct manner and require substantial R&D work. The European community decided to launch the High Performance Light Water Reactor (HPLWR) project, which is now completed, to check the feasibility of the super-critical water cooled reactor concept.*

The starting point for such studies uses an advanced design by Dobashi et al, [1] for its comprehensive definition of all the relevant aspects of a HPLWR design. During the course of the HPLWR project, an extensive evaluation of core characteristics has been made and alternative options studied.

This "reference design" has interesting and innovative features such as water rods which help to flatten the axial power shape as this is one of the main disadvantages of a HPLWR core and also helps to compensate the reactivity swing. However, the "reference design" has features such as a relative small moderation, a moderation very much dependent of the magnitude (and amplitude) of the variation of water flows and of their thermal insulation, a non-uniform water flows in the different sub-channel of the sub assembly and the use of Ni-based alloy cladding materials with a relative strong neutron absorption.

With various and appropriate design changes which have been studied during the course of this HPLWR project, the HPLWR "reference concept" could be improved but this, however, would require a substantial design effort with all constraints taken into account. Rather complete new revolutionary designs could also be envisaged but this would even increase the overall design effort. At the conclusions of the HPLWR project, there is a much clearer understanding of what constraints apply to the design, and what is required to upgrade the necessary design tools to pursue the analyses, supported by experiments, within the 6th FP.

I. INTRODUCTION

The super-critical pressure light water reactors present some advantages over existing reactors (PWR and BWR) which prompted the European community (within the 5th FP) to study such a possibility. Within the High Performance Light Water Reactor (HPLWR) project , the objectives of the specific group on core design were:

- To evaluate existing core design proposals
- to suggest potential solutions for a fuel assembly and control rods, and
- to identify code requirements for neutronics and thermal-hydraulics for the HPLWR with thermal spectrum.

In order to achieve these objectives, an advanced design by Dobashi et al. [1] has been chosen for its comprehensive definition of all the relevant aspects of a HPLWR design. This design taken as HPLWR "reference concept" is described in the following chapter (Chapter II) using tools which have been extensively studied (Chapter III). The evaluation of the HPLWR "reference concept" has been evaluated during the course of this project (Chapter IV) and potential solutions for improving its design or even drawing a complete new innovative design have been investigated (Chapter V). The conclusion is providing some guidelines for next phases of HPLWR design projects (Chapter VI).

II. THE HPLWR "REFERENCE CONCEPT"

The thermodynamic cycle of a reactor cooled and moderated with supercritical water is very similar to those of fossil power plant (FPP) using similar coolant. It uses a direct cycle and the cooling circuit is operating at a pressure of 25.0MPa (supercritical fossil power plant reach even up to 30.0 MPa). As the water flow required to cool the core is relatively small, moderation is not sufficient and in the HPLWR "reference concept", the thermal spectrum is provided by water flowing downwards in dedicated water rods. Input and output temperatures are assumed at 280°C and 508°C respectively. These temperatures lie on both sides of the pseudo-critical temperature (~385°C).
The fuel subassembly has numerous water rods in order to enhance the core moderation.
The fuel sub-assembly of the "reference design" contains 96 rods with low average enrichment (4.16%), 48 rods with medium average enrichment (5.16%) and 114 rods with high average enrichment (6.0%) that are arranged in 12 rows and with the enrichment increasing from the inside to the outside of the fuel sub-assembly. This model results in an average fuel assembly enrichment of 5.16%.

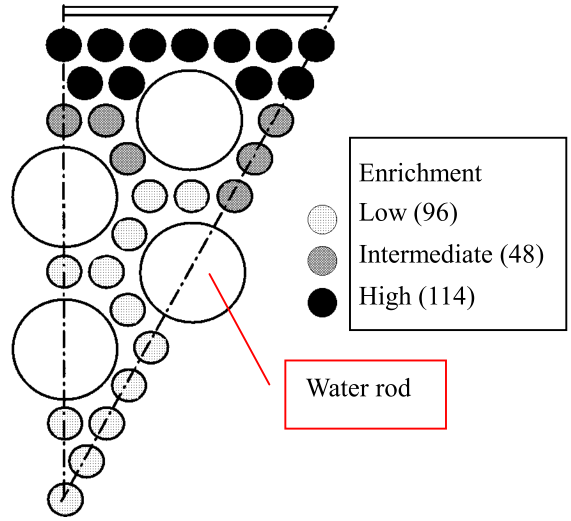

Figure 1 – Simplified subassembly of the HPLWR "reference concept" by Dobashi et al. [1] and Pin Enrichments Within The Fuel Sub-assembly

The pin diameter is 0.80cm, the cladding is 0.040cm thick Nickel alloy, the distance between two fuel pins is of 0.95cm while the subassembly pitch is of 21.5cm. The sub-assembly wrapper is also made of Nickel alloy and is 0.15cm thick. The purpose of such a wrapper is to allow different coolant water flows in the sub-assemblies and therefore, given the high temperature increase in the core, avoiding important thermal losses. Average Uranium enrichments and Gadolinium concentrations have different values axially and in the radial direction in order to better distribute the power in the core.

In the HPLWR "reference concept", the water flow in the rods is modified during the operation of the plant in order to compensate part of the reactivity swing. This feature is using the fact that these water rods are not fully insulated. A high water flow will absorb less heat in the water and moderation remains then high along the core moderation which therefore induces a high reactivity specifically required at the end of cycle (EOC). A low water flow in the water rods will increase the temperature of the water, therefore reducing the moderation and this is required at the beginning of the cycle (BOC). Some kind of limited water rod insulation is provided by stagnant water around the water rod. A thermal-hydraulic study on the thermal exchange between coolant water and the water rods has been performed, indicating the importance of the thermal insulation of the water rods.

III. APPLICABLE NEUTRONICS/THERMAL-HYDRAULICS CODES, CROSS SECTION DATA BASE AND NEUTRONICS TEST REQUIREMENTS

III.A. Evaluation methodology of existing codes and data

Defining a calculation scheme suitable for designing HPLWR concepts without appropriate verifications might lead to uncertain results. A few verifications are therefore required, those related to methods with the help of a Monte Carlo code while those related to nuclear data are checked by an interchange of nuclear data libraries.

The calculations performed within the framework of the HPLWR project included: 2-D sub-assembly cell calculations, 3-D hexagonal core calculation for a limited number of groups and including burnup calculations, thermal hydraulics calculations including iterations with 3-D hexagonal core calculation, reactivity changes over time with burnup and reactivity coefficients calculations.

Tools available in Europe are very similar to those used by the University of Tokyo with the exception of the thermal hydraulic conditions of the water rods which has been introduced by incorporating the SPROD code of the University of Tokyo.

The parametrized cross section library has been built into the GLOBUS nodal code of the KARATE program system [3] for calculating core characteristics. Also ERANOS [4], KARBUS and MCNP[5] have been used during the course of this project with libraries of ENDFB VI and JEF2.2 origins.

III.B. 2-D sub-assembly cell calculations

Six cases were defined in order to cover most thermal-hydraulics conditions and in order to distinguish between the effect of enrichment and the effect of water density. Furthermore, the water temperatures (and densities) were selected on each side of the pseudo-critical point. The fuel cladding, the wrapper tube, the water rods and the stagnant water were assumed to be at the coolant temperature (350 ºC), the fuel temperature was assumed at 1227 ºC and the fuel oxygen stochiometry was assumed at 1.98 (O/U). The six cases are then: Cases 1- 330 ºC, 0.6807 g/cm3, 4.72%, (water temperature, water density, average fuel enrichment), Case 2- 370 ºC, 0.5405 g/cm3, 4.72%, Case 3- 370 ºC, 0.5405 g/cm3, 5.16%, Case 4- 400 ºC, 0.1665 g/cm3, 5.16%, Case 5- 400 ºC, 0.1665 g/cm3, 6.0%, Case 6- 480 ºC, 0.0960 g/cm3, 6.0%.

The results for Keff were compared for the six cases between ERANOS (using the JEF2.2 library and the adjusted ERALIB1 library), MULTICELL (using ENDFB VI.3 library). and MCNP (using both JEF2.2 library and ENDFB VI.3 library).

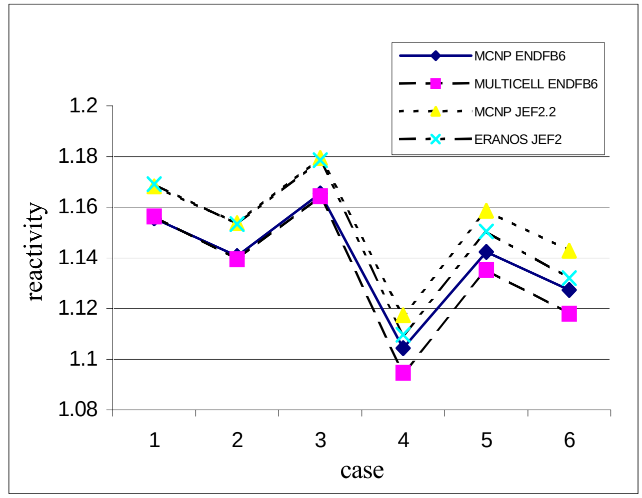

Figure 2 - Comparison Between Benchmark Results

The agreement between the three codes is within 2%. Different nuclear data libraries have been used and this explains most of the discrepancies. Existing experiments for assessing the nuclear data and computer codes are available for PWR or BWR UO_2 fuels; for different moderation ratios and different parameters (initial reactivity, power distribution, material balance in burnup calculations, control rod reactivity worth, feedback reactivity worth, i.e. Doppler, temperature, void, and kinetic parameters). Current analyses (of EOLE and PROTEUS facilities) show that JEF2.2 over-estimate the experimental reactivity results whereas ENDFB6 underestimate them.

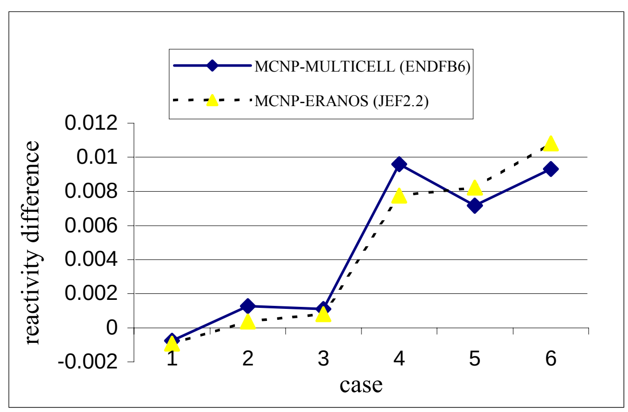

Figure 3 - Comparison Between Benchmark Results for Reactivity Differences of Deterministic and Monte Carlo Codes Using Similar Nuclear Data Sets

The results are affected by the fact that current nuclear data libraries have hydrogen cross sections tabulated up to 350°C in both JEF2.2 and ENDFB6 ones either for deterministic codes or for the Monte Carlo code MCNP.

But MCNP and deterministic codes use Hydrogen temperature dependent cross sections differently, MCNP uses the highest tabulated cross section for cases with high temperatures (but with the density corresponding to this temperatures) while ERANOS and MULTICELL deterministic codes use an extrapolation methodology which looks better but might be unreliable. The correct method, however, is to use an interpolation technique on α and β before getting the S(α, β) value which is used to get the thermal matrices. Experimental validation to check that this is correct is not available above 550 ºK. It is therefore suggested to set up an experimental programme at the Laue Langevin Institute to get the appropriate information on the S(α, β) values for Hydrogen bound in water for temperatures higher than 350°C just like it has been done for zirconium hydride or calcium hydride recently but in another context [6].

In this analysis, 2D slice subassembly calculations at BOL were also performed for reactivity coefficients (thermal expansion, Doppler, water density, water rod density). Thermal expansion was assessed from cold coolant temperature to the average coolant temperature and for Iron or Ni alloy, Doppler was assessed for cold fuel (293ºK) while other regions of the assembly were kept at the same temperature, water density was assessed for a 1% decrease in the density of the coolant water and for completely voided assembly (99% decrease in coolant density), water rod density coefficient was assessed for the water rods being at the same temperature as the water coolant.

ERANOS results of the calculations are as follows: for the Doppler reactivity coefficient when the fuel temperature is changed to 293ºK while other regions are kept at the same temperature and compared with Keff without thermal expansion, the coefficients vary for the six cases between –2.36 pcm/ºC (for case 1) and –3.04 pcm/ºC (for case 6); water density coefficients for a 1% decrease in water coolant density vary between –0.0909 (for case 1) and –0.2453 (for case 6), and for 99% decrease in the coolant density (voided assembly) the density coefficients vary between –0.1732 (case 1) and –0.2598 (case 6); the water rods density coefficients (when the rods are at the temperature of the coolant) vary between –0.0678 (for case 1) to –0.4620 (for case 6). Similar values were calculated with both the deterministic MULTICELL code and the MCNP Monte Carlo code. For Doppler coefficients no discrepancies between results using different codes and data appeared while void coefficient results were different from each other, but with MCNP results being close to that of ERANOS. One could suspect in this case method difficulties associated with the MULTICELL code. Experiments in zero power facilities would once again help in finding the right value.

Comparison between the results of different deterministic codes was also done for different burnups; Keff was calculated for the six cases at seven different burnup steps (0, 220, 440, 660, 880, 1100, 1320 FEPD-Full Effective Power Days), assuming a water rod temperature of 350 ºC (0.6250 g/cm^3). The core characteristics of the “reference design” had a power density of 101 W/cm^3, a discharge burnup of 45 GWd/t and a refuelling period of 440 days. Cycle step reactivities were calculated as average of the sub-assembly step reactivities assuming that 1/3 of the reactivity is due to fresh fuel, 1/3 from 440 FEPD and 1/3 from 880 FEPD irradiated fuel. The different slice values were further averaged axially. This rough estimate of course does not substitute the real core burnup calculations but are sufficient for understanding the possible bias coming from codes and nuclear data. The Kinf of the assemblies were calculated by the MULTICELL and the ERANOS codes on these 440 day steps:

Table 1 - Kinf Results by MULTICELL and ERANOS

Enrichment	Low	Medium	Medium	High
Coolant Temp.	330°C	370°C	400°C	480°C
MULTICELL	case1	case3	case4	case6
BOC	0.99237	1.00693	0.95888	0.98268
MOC	0.91510	0.93330	0.89750	0.92130
EOC	0.85493	0.87723	0.85513	0.87874
ERANOS	case1	case3	case4	case6
BOC	1.02005	1.03462	0.98520	1.00704
MOC	0.94625	0.96431	0.92666	0.94911
EOC	0.89037	0.91223	0.88765	0.90979

(BOC -Beginning of Cycle, MOC- Middle of Cycle, EOC - End of Cycle)

Initial reactivity values differ by approximately 2% and this, as previously mentioned, has most probably a nuclear data origin, but the reactivity swing is much sharper with the MULTICELL code than with ERANOS. The source of these differences is most probably coming from nuclear data (ENDFB6 and JEF2) and should be investigated as it impacts the core design significantly. There are actions within the European JEFF project to clarify this issue which is also a matter of concern for designs or other reactor types (PWR or BWR).

These burnup calculations showed that the reactivity swing could not be compensated by control rods, that Gadolinium should be introduced in some fuel pins, and that the water rod density should be changed during the cycle.

III.C. Sub channel Analysis

In order to perform the analysis for the HPLWR, data of supercritical water as well as heat transfer correlations and a correlation for deteriorated heat flux were added to the FLICA code [7]. The “reference design” was then analyzed with the modified FLICA code while assuming a cosine axial power. The results indicated substantial impact of the heat transfer coefficient on the axial temperature profile of the cladding, with the Dittus-Boelter correlation yielding the highest cladding temperature. Calculations of the temperatures within the "reference fuel assembly" showed large temperature differences (200 ºC) between the center of the assembly and the temperature around the water rod. The reason for the large temperature differences is the peculiar geometry of the water rod wrapper that generates regions of low velocity. The results also indicate the radial variations of the axial velocities within the assembly, that indicated regions of high and low axial velocities (up to a factor of 3 difference) which in turn cause high temperature variations in the assembly. Similar calculations were performed by Cheng et al. [8]. They showed that in some parts of the sub-assembly, the cladding temperature exceeds the 620°C cladding temperature criterion. This is due to a too heterogeneous distribution of the pressure losses in the sub-assembly.

A second set of calculations by Cheng et al. [8] was done on the effect of thermal insulation of the moderator rods under the assumption that 47% of the water is flowing downwards through the water rods. The analysis was done by assuming a high heat transfer coefficient (40 kW/m2K) for conducting walls, a low value (1 W/m2K) for insulating walls, and 1000 W/m2K for the "reference case". A very significant density reduction of the water rods was calculated for conducting walls (no insulation).
For the "reference case" the coolant heats up from about 310°C at the core inlet to 508°C at the outlet. Thus the average water density in the core, including the water in the moderator rods as well as the coolant water, decreases from 690 kg/m3 at the inlet to a minimum of about 450 kg/m3 near the core outlet. For the case of conducting walls, the coolant temperature at the core inlet is increased to about 350°C but the coolant temperature reaches only about 500°C at the outlet of the sub-channels which is not adjacent to a water rod, which is slightly lower than the reference case..
The average water density decreases from only 500 kg/m3 at the core inlet to a minimum of about 280 kg/m3 at the middle of the core. This could be a concern to the core moderation and the reactivity coefficients.
For the case of perfectly insulated water rods, the outlet coolant temperature for the sub-channel next to a water rod reaches about 620 ºC so that the cladding temperature constraint of the "reference design" will clearly be exceeded. The core-average water density decreases for this case from about 780 kg/m3 at the core inlet to about 450 kg/m3 at the core outlet. The large variation of water density in the core can be reduced if 90% (instead of 47%) flows downward through the water rods. Under this assumption and for the case of conducting walls (without thermal insulation of the water rods) the core-average water density varies from 530 kg/m3 at the core inlet to about 380 kg/m3 at the middle of the core height. However, the average water density for this case is still significantly lower (by about 150 kg/m3) than the "reference case".
These results indicate that:

(a) a sub-channel analysis and verification is needed for a detailed core design,
(b) coupled neutronics/thermal-hydraulics analysis is necessary,
(c) a simpler core design, without the need for thermal insulation is preferable.

The above mentioned sub-channel analyses have highlighted some of the shortcoming of the fuel assembly of the "reference design"and have also indicated that the sub-channel codes are very helpful and mandatory for a detailed HPLWR core design.

III.D. 3-D core calculations

A parametrized 2-group cross section library for low burnup values has been generated (parameter range covers the cold zero power and hot full power states) for use by the GLOBUS nodal code of the KARATE program system and with neutronics/thermal-hydraulics coupling using the University of Tokyo's SPROD code.
The BOL reactivity is significantly lower than the one expected. This could come from the material used for the reflector. Another material could raise significantly the initial reactivity. The following values of Keff were calculated: 1.05684 for unburnt core, hot, full power, equilibrium Xe, and 1.25819 for unburnt core, cold, zero power, zero Xe. Extensive analysis of the HPLWR core has been performed while varying the ratio of water rods flow to the total flow between 0.05 and 0.60.

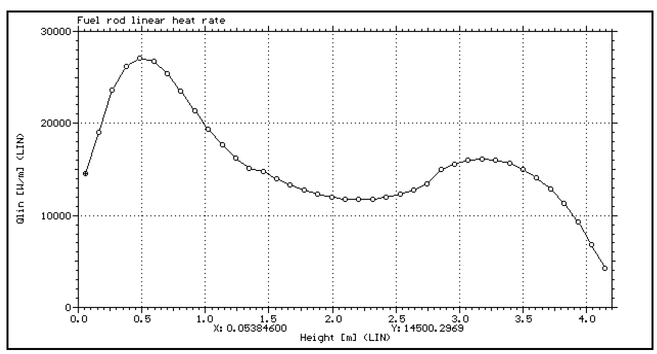

Figure 4 - KARATE Calculated Axial Power in The Fuel Assembly

The axial power profile in an average assembly with water rods is shown in Figure 4 and differs quite significantly from the original curve shown by Dobashi et al [1] but these were for an equilibrium cycle while the illustrated

KARATE distribution corresponds to fresh fuel without Gd burnable absorber using equal-distant axial enrichment zoning.
The analysis showed that the axial power profile does not follow a cosine profile but rather has a shape with two peaks, the location of which changes with the flow rate. The calculated reactivity coefficients show sufficient stability against inlet temperature and fuel temperature rise (-2.35 pcm/K for the fuel and –38.7 pcm/K for inlet temperature).
With the KARATE code system, the effect of Gd burnable absorbers on reactivity swing was studied by performing sub-assembly burnup calculations of the "reference design" with the MULTICELL code. The analysis was done for the same six axial assembly slices that were defined by the 2D sub-assembly benchmark problem. The burnup calculations were presented as a function of Full Effective Power Days (FEPD) and a 440 day equilibrium cycle. The results of these calculations indicate that for the "reference design" the burnout of Gd overcompensate the reactivity loss of fuel burnup and therefore a smaller number of Gd rods should be used. As a rough estimate 4% of reactivity loss should be compensated between MOC and EOC when the Gd content is unchanged. This analysis also shows that the axial power profile does have a shape with two peaks (see KARATE power profile in Figure 4 above), the location of which changes with the flow rate in the water rods and with the introduction of Gd rods (54 Gd rods were used in the analysis). Based on this analysis, it is recommended that once the fuel assembly design has been fixed, a core benchmark analysis be carried out together with a parametric and sensitivity study.

III.E. Experimental validation

With respect to experimental validation of the neutronics analyses, there are some existing relevant data at the beginning of life (BOL) from previous experimental programs (e.g. EOLE, VENUS, PROTEUS, etc.) for both UO_2 and for MOX as well as irradiation data. After the fuel assembly design has been finalized, it is recommended to perform validation experiments, in particular power map distribution and reactivity worth. The experiments could determine the pin power, reaction rate and relative reactivity effects on neutron absorbers in a mockup fuel assembly of the HPLWR and could be performed in zero power facilities such as PROTEUS [9] or EOLE [10].
At this moment, these conclusions stand only for UO_2 fuels but one could expect that a similar validation work is required for MOX fuels. Finally, a fast HPLWR core would require more specific numerical and experimental validations.

III.F. Conclusion of existing codes and data for HPLWR applications

The HPLWR benchmark study demonstrates that the codes give reasonable results but the nuclear data need improvements before making the final calculations; experiments should be performed at high temperatures (350 ºC to 600 ºC) to determine the impact of bound effects of Hydrogen within water or other hydride moderator materials; Hydrogen tabulation above 350 ºC should be produced, whatever the experimental situation for bound effects is; study of the predictability of the nuclear data and code systems should be done on existing experiments; experiments should be performed on the final sub-assembly design. Nevertheless, the current predictability of the nuclear data and code systems is adequate for preliminary design.
These computational tools have to be improved for advanced studies. Improvements can be achieved by: comparison of codes and data, comparison of data sets, experiments to be analysed, other experiments to be set up in experimental facilities such as EOLE [7] or PROTEUS [8].
It is necessary to use coupled neutronics/thermal-hydraulics codes due to the strong coupling between neutronics and thermal hydraulics in a typical HPLWR core.

IV EVALUATION OF THE HPLWR "REFERENCE CONCEPT"

IV-A Introduction to the evaluation

The HPLWR "reference concept" has demonstrated its desirable features, in particular its high thermal efficiency and the possibility to control reactivity swing by changing the flow in the water rods during the cycle. However several of its characteristics look unconventional such as:

- Under-moderation leading to a large reactivity swing to be compensated during the cycle by complicated systems (e.g. change in descending water flow during the cycle),
- Fuel enrichment that could reach 7 %, that is well above the licensing limit of currently operating commercial enrichment facilities (5 or 5.5%),
- Three different enrichment zones of the fuel pins within the sub-assembly.

Consequently, it was decided to find out the reasons for such "reference design" characteristics. Analyses of the "reference design" have then been performed with simple calculations limiting the study to neutronic characteristics for given thermal-hydraulic conditions. The study of the HPLWR design includes in particular a detailed neutronic balance, an evaluation of the moderation that can be achieved and a distribution of this moderation within the sub-assembly.

IV-B Core moderation

In order to evaluate the moderation ratio distribution, five rings have been defined. In each of these rings the following ratio H/N has been calculated:

$$H/N = \frac{\text{Nb of atoms of Hydrogen}/cm^3}{\text{Nb of atoms of Heavy Nuclides}/cm^3}$$

With moderator water temperature set to 350°C, the values of the moderation ratios are plotted in figure 5.

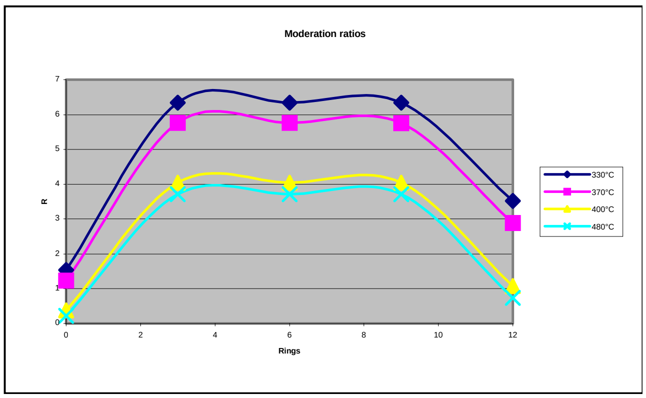

Figure 5: Moderation ratios as a function of the ring position and coolant temperatures.

The shape of the curves reveals that water rods contribute significantly to the moderation.

- Between 370°C and 400°C (the value of the pseudo-critical temperature is 385ºC) there is a drop in the water density and therefore of the moderation ratio.
- The water regions on each side of the wrapper of the subassembly create a better moderation in the peripheral region than in the central region.

For calculating the average core moderation, an enthalpy balance is performed in each of 20 elementary axial regions assuming a cosine axial power distribution and 280°C for the inlet temperature. The value is found to be 3.66 which is far below the one of standard PWR.

Table 2 - Moderation in comparison with existing concepts

Concepts		HPLWR	PWR	Under-moderated reactor (MOX)	
				original	advanced
Moderation	$\frac{H}{HN}$	3.7	4.5	3.2	1.6
	$\frac{V_{H_2O}}{V_{FUEL}}$	2.58	2	1.4	0.7

Although the volume moderation ratio of the HPLWR is higher than that of the PWR, moderation in the current HPLWR concept is insufficient since it is the H/HN ratio that determines the moderation. Under-moderated reactors when using MOX fuels were designed to increase the conversion factor, i.e. minimize the Pu consumption. In that sense the HPLWR "reference core" meets the corresponding objectives which are a path towards a sustainable source of energy. More work is required to verify this is possible although preliminary work on reactivity coefficients are encouraging. These under-moderated cores using UO_2 fuel are not optimized as their balance include a relative high absorption of H_2O.

IV-C Neutron Balance

The neutronic balance has been studied for the following cell (coolant temperature:370°C, coolant density: 0.5405 g/cm^3, average enrichment: 5.16%) for which K_{eff} =1.18711 with ERANOS.

In table 3, the leakage for this thermal cell is rather important (16%) which is the consequence of its relatively low moderation. Ni-alloy contribution to the capture is significant (12% over a total capture of 43%) and the choice of using such material is associated to its behavior under HPLWR operating conditions.

Table 3 - Neutronic Balance of the Average Cell

Production	100%		
Leakage	16%		
Fission	41%		
Capture	43%	Fuel	28%
		Ni-Alloy	12%
		Water	3%

The use of this very absorbing material together with the under-moderation requires a significant increase of the fuel enrichment.

IV-D Enrichment distribution

Independently from the enrichment level, the distribution of the enrichment in the sub-assembly is set to reduce the radial peaking factor significantly as it has been demonstrated using MCNP. The reason for having three enrichment zones has been tracked back in the same manner as the distribution of moderation. It has been found that the absorption distribution in the sub-assembly is quite affected by the existence of the wrapper tube (made of Ni-Alloy) and that effect is contributing in the same direction than the moderation to the distortion of the spatial neutron balance of the sub-assembly. The three enrichment zones {average value 4.72% (three pin enrichments 3.81%, 4.72%, 5.49%), average value 5.16% (three pin enrichments 4.16%, 5.16%, 6.00%), average value 6.0% (three pin enrichments 4.87%, 6.04%, 7.02%)} are compensating these effects. This analysis if of course at BOL and in order to achieve similar results over the life of the sub assembly Gadolinium pins are introduced.
In order to reduce the fuel enrichment that reaches 7 %, well above the licensing limit of currently operating commercial enrichment facilities (5 or 5.5%), several actions could be performed such as the increase of the subassembly gap size, the use of austenitic stainless steel instead of Ni-alloy and eventually the use of hydride pins.

V STUDIES IN SUPPORT TO IMPROVED OR ALTERNATIVE DESIGNS

V-A Introduction to the studies

Guidelines for improved design are supported by an investigation of various options aimed at improving the HPLWR "reference design":

- change in the structural material (for both cladding and wrapper),
- change in the moderation with the use of:
  - Hydrides (YH_2 and $ZrH_{1.78}$) rather than water rods
  - Water-gap between subassemblies: larger and/or filled with cold water

But, when introducing any change either small or big in the design, constraints which apply to it have to be taken into account and in order to better prepare the next phases of the design project, they have been reviewed in a dedicated chapter.

V-B Structural materials

In the "reference design", Ni-alloy has been proposed (Inconel 625). In the frame of the HPLWR, it has been recommended that stainless steels could also be used and are preferred over Ni-based alloys. From the neutronic design viewpoint, this is quite important. The variation of absorption with material types has been analyzed calculating Keff for the same material thickness. For 1.4970 (a typical stainless steel (SS)) the value of Keff is 1.14811 and for Inconel 718 (Ni-based alloy) Keff is 1.09750. The contribution of Ni absorption to the total neutron absorption is 14% and is reduced to 10% with SS. Another way to present that is to say that the need for an average enrichment is decreased by 0.90% when SS is used in the core instead of Ni-based alloy.
An analysis of the required cladding thickness was performed. Considering 100 and 200 MPa hoop stress, the maximum material temperatures were determined for a creep exposition to these loads during 45000 hours of in-core service.
For the austenitic stainless steel 1.4970, the maximum temperature is 690°C for 100 MPa and 629ºC for 200 MPa. The cladding thickness for an 8 mm fuel pin diameter would be 0.68 mm for 100 MPa, 0.45 mm for 150 MPa and 0.34 mm for 200 MPa [8]. To obtain the same 690°C maximum temperature as 1.4970 SS, the hoop stress for Inconel 718 can be increased from 100 MPa to about 160 MPa. This means that the wall thickness for Inconel 718 can be diminished to about 0.45 mm.
The proposed dimensions of the wrapper could be changed accordingly with a similar relation, although the operating temperature for the wrapper is much smaller and other criteria than creep properties would be relevant.
With a maximum operational temperature of 650°C, one can use austenitic steel 1.4970 at a stress level of 160 MPa with the same cladding dimensions as with Inconel at 690 °C. A stress level of much more than about 150 MPa could introduce other problems, such as the buckling of thin-walled tubes that could occur under these conditions.
The observed higher sensitivity of Ni-alloys to stress corrosion cracking and irradiation-induced helium embrittlement are also important arguments in favor of stainless steels up to about 650°C.

Therefore, only if cores are designed for maximum cladding temperatures superior to 650°C in normal operation is the use of Ni-alloys necessary.
But based on a comprehensive analysis, taking into account numerous factors including creep, fission gas pressure, corrosion, neutron damage, helium production, etc. austenitic stainless steel has been recommended for in-vessel material.
If there is a need to increase the thickness of in-vessel material, this increase will be accompanied by higher enrichment due to higher neutron absorption by the material and it is therefore recommended that these aspects should be taken as important constraints in the future design of the HPLWR subassembly.

V-C Hydride materials

Despite the advantages of the water rods (simplicity and reduction of specific radioactive waste), they set some design and water insulation problems whereas a solid moderator is simpler to design and its moderation is independent of temperature. The possibility of using solid hydrides could be envisaged provided that their thermo-chemical and mechanical properties at high temperatures can cope with the HPLWR thermal-hydraulic conditions.
In order to investigate the possible use of these hydrides materials, calculations have been performed by replacing the water rods with these materials (YH_2 and $ZrH_{1.78}$). Table 4 demonstrates that hydrides induce a penalty of the reactivity due to their higher absorption.

Table 4 - Comparison between hydrides & water rods with a 370°C coolant water

Structure material	YH_2	$ZrH_{1.78}$	Water rods
K_{eff}	1.14071	1.17932	1.18711
Number of hydrogen atoms (10^{24}atoms/cm^3)	5.68 10^{-2}	5.02 10^{-2}	4.18 10^{-2}
Capture rates/cm^3	6.32.10^{-4}	3.24.10^{-4}	1.69. 10^{-4}

It can be seen from Table 5 that the Ytrium is the most capturing material but:

- the neutronic balance changes between the different cases studied only due to rod capture and leakage
- $ZrH_{1.78}$ rod is not much more capturing than the water rod.

Table 5 - Capture break down comparison

	YH_2	$ZrH_{1.78}$	Water rods
Production	100		
Leakage	12	15	16
Fuel capture	28		
Ni-Alloy capture	12	12	12
Moderating rod capture	7	4	3
Fuel fission	41		

By keeping the same temperature repartition along the core for YH_2, $ZrH_{1.78}$ and water rods, the average core moderation has been calculated and gives values of respectively 4.34, 4.04 and 3.66. Moderation increases by using YH_2 rods instead of water rods, the reason being a reduced leakage that is only counterbalanced by the larger capture of the Ytrium. From the neutronic point of view, zirconium hydride rod is the best one. Yet, its residual radioactivity after irradiation and its high dissociation pressure remain important disadvantages especially during reactor transients. YH_2 is a more stable hydride and could be a better candidate.

V-D Water gaps between subassemblies

In Table 6, results of investigations associated to the possibility of enhancing the moderation by increasing the size of the gaps between subassemblies are presented.

Table 6 – Moderation change due to S/A gap

	S/A gap size	Water temperature (°C)	Average core moderation
Reference Case	2 mm	Coolant	3.66
Case with cold water	2 mm	350°C	3.75
Case with hot water	11 mm	Coolant	4.06
Case with cold water	11 mm	350°C	4.34

Only a low water temperature in a 11mm thick gap can increase the core moderation to levels almost equals to the PWR core moderation of 4.5. The increase of the water gap has also favorable consequences on the power distribution as moderation is increased in the boundary of the sub assembly.
The feasibility of such a feature is associated to the water flows which are required by cooling of reflector and RPV closure and are drained through the core in downward flow and fed into the fuel assembly entrance. Part of this cold water flow could go through the gap between two adjacent fuel sub assemblies.

V-E Design constraints and guidelines for future designs

It is not the purpose of this paper to present the overall constraints which apply to the design of an HPLWR plant but only to review those constraints which have a direct impact on the design of a core and hence on its efficiency.
In current LWR's, the safety requirements are respected using two core parameters : the linear heat rate and the DNBR (Departure from Nucleate Boiling Ratio). Taking into account some margins on these parameters of heat transfer deterioration during the transients, the nominal

core design parameters are about 2.8 for the DNBR and 400 W/cm for the linear heat rate.

Given that the heat transfer deterioration in super-critical water is much milder than the burn-out (or Critical Heat Flux) at sub-critical pressures, a criterion based on the maximum cladding temperature is now considered [11,12]. The nominal core design parameters obtained are then (for a Ni based alloy):

- maximum linear heat rate of 390 W/cm,
- maximum cladding temperature of 620°C.

It must be recalled that the heat transfer to supercritical water in a bundle type geometry is not well known to allow the accurate calculation of the cladding temperature and this key missing information would require experimental validation. Having a once-through core cooling and given the core radial power profile, it is necessary to distribute the coolant flow rate as a function of the radial position of the assemblies. Therefore, enclosed assembly design is required. Such solution could also limit potential hydraulic instabilities.

In order to cool the reactor vessel and some internal structures (core reflector), some minimum bypass flows are required for the reactor vessel head and the core radial reflector. In the HPLWR "reference design", the water flow cooling the vessel head is also used to fill the water rods. This is of course quite attractive solution to avoid to mix this flow with the core outlet flow which would then decrease the "steam" output temperature. In addition, in this design, this flow is also modified over time in order to modify the reactivity and therefore to compensate part of the large reactivity swing. The range in which one could modify these water flows in the course of reactor operation is a matter of concern and still not solved at the end of this project.

Extra water is added in the assembly in dedicated tubes (water rods) in order to enhance the moderation. Due to the heating of the water in these dedicated water rods (whether they are insulated or not), the water density is decreased while flowing in the core. In order to compensate for the axially distorted power shape in this type of core, there is a preference to have a descending water flow within the water rods. With such a design, the mixing of water rod flow with the outlet steam flow is avoided. This arrangement should be compatible with the fact that the fuel sub-assembly (S/A) should be removable from the core. This feature is quite challenging.

The water rods are surrounded by stagnant water in the "reference design" in order to insulate the water inside these rods from the core coolant. A double tube is proposed. It may not be necessary to insulate that water if the downward water flow rate is sufficiently large. A specific study with larger water rods [13] has shown that without insulation, the coolant temperature in the water rod is below the pseudo-critical temperature (i.e. higher water density is available for moderation) and that this result is not very sensitive to the flow rate in the water rods for flow fraction larger than ~30%.

In addition, the mechanical arrangement is somehow complicated by the existence of this double tube, therefore the removal of the second tube and substituting it with a single tube is being considered. However, it is possible that the differential temperature across the tube wall may lead to significant thermal stresses and thermal-mechanical constraints. Additional studies including experiments are necessary to resolve these issues.

The finding of new sub assembly drawings and associated core meeting all design constraints is a challenging new optimization problem to be solved and would require a substantial increase in effort compared to the current project. The decreasing moderator power of the supercritical water in the upper part of the core is to be compensated by solid moderator, like $ZrH_{1.8}$, YH_2 or by water gaps between subassemblies. Also the consequences for the thermal-hydraulic overall design have to be analyzed. In the case of MOX fuel, under moderated regular water lattices have an epithermal neutron spectrum with good conversion ratio of fertile into fissile fuel and small burnup reactivity loss. This option is quite attractive from what has been done within the project but might not meet some of the GEN-IV criteria such as sustainability. It is therefore recommended to look also for a fast version of a supercritical water core.

VI. CONCLUSIONS

Some progresses were made during the HPLWR project of the 5th FP with respect to the applicable neutronics/thermal-hydraulics codes, cross section date bases and neutron physics test requirements. The calculations included: 2-D sub-assembly cell calculations, 3-D hexagonal core calculation for a limited number of groups and including burnup calculations, thermal hydraulics calculations including iterations with 3-D hexagonal core calculation, reactivity changes over time with burnup and reactivity coefficients calculations. For the 2-D subassembly analysis the agreement between k-infinity results was within 2% which is significant, and it was concluded that the nuclear data is the main source of these differences. However, such differences exist also when calculating PWR and the need for significant improvements on nuclear data is shared over the reactor physics community and for various reactor applications. More specifically to HPLWR, data on bound hydrogen in water do not exist above 350 ºC in current libraries, it should be generated by theoretical analysis and validated by specific measurements at the ILL facility in Grenoble (France).

Extensive analysis of the HPLWR core were performed while varying the ratio of water rods flow to the total flow between 0.05 and 0.60 and with neutronics and thermal

hydraulics coupling using the University of Tokyo's SPROD code. This analysis showed that the axial power profile does not follow a cosine profile but rather has a shape with two peaks, the location of which changes with the flow rate. Based on this analysis, it is recommended that once the fuel assembly design has been fixed, a core benchmark analysis be carried out together with a parametric and sensitivity study. With respect to experimental validation of the neutronics analyses, there are some existing relevant data at the beginning of life (BOL) from previous experimental programs (e.g. EOLE, VENUS, PROTEUS, etc.) for both UO2 and for MOX as well as irradiation data.
After the fuel assembly design has been finalized, it is recommended to perform validation experiments, in particular power map distribution and reactivity worth. The experiments could determine the pin power, reaction rate and relative reactivity effects on neutron absorbers in a mockup fuel assembly of the HPLWR and would be performed in zero power facilities such as PROTEUS or EOLE. It was concluded that the available analytical tools are adequate for pre-design studies, however they must be improved for a more advanced and detailed design. Improvements can be achieved by comparing codes and data, comparing data sets, experiments to be analyzed and additional experiments to be performed. Also, it is necessary to use coupled neutronics/thermal-hydraulics codes because of the strong coupling between neutronics and thermal-hydraulics in a typical HPLWR core.

Therefore, the new and unconventional design features of the HPLWR cores are requiring R&D work which can be summarized in the following:

- Data on bound hydrogen in water should undergo theoretical analysis and validation on specific measurements at the ILL facility in Grenoble (France).
- Differences on reactivity values coming from nuclear data (ENDFB6 and JEF2) should be investigated within the European JEFF project. This is also a matter of concern for designs of other reactor types (PWR or BWR).
- A preliminary assessment of computational tools is possible using existing relevant data at the beginning of life (BOL) from previous experimental programs (e.g. EOLE, VENUS, PROTEUS, etc.) for both UO2 and for MOX as well as irradiation data.
- However, benchmark studies indicate the necessity for performing experiments that could investigate the effects of different moderation conditions in the water density range of interest to the HPLWR (0.2÷0.7 g/cm3 in the fuel assembly, 0.7 g/cm3 in the water rods).
- Once the fuel assembly design has been fixed, computational tools would have to be validated on experiments for core characteristics, such as pin power, power map distribution and reactivity worth, performed in zero power facilities such as PROTEUS or EOLE. These conclusions stand not only for UO2 fuels but also for MOX fuels.
- A fast HPLWR core would require more specific numerical and experimental validations.
- On the thermal hydraulic side, basic data on heat transfer, pressure drop and critical flow for supercritical water in a bundle type geometry would have to be acquired via experimental set up.
- The analysis of the HPLWR core with neutronics and thermal hydraulics coupling for different ratios of water rods flow to the total flow would require numerical validation. This validation would concern the axial power profile and reactivity feedback coefficients and could be done together with a parametric and sensitivity study.

After evaluating the "reference design" it was concluded that the core is under-moderated and uses excessive neutron-capturing structural material (Ni-based alloy) as well as non-uniform sub-channel flow, thus imposing a substantial penalty on this concept. The under-moderation leads to a large reactivity swing that should be compensated and the fuel enrichment can reach 7%, which is well above currently operating commercial fuel production facilities. Also, the different enrichments of the fuel pins within the sub-assembly leads to some complications and the original design burn-up of 45 GWd/t is too low (proposal is to increase it to 60 GWd/t therefore reducing significantly the fuel cost penalties). Additional effort is needed to evaluate MOX fuel cores, plutonium use in the core, and fast cores. Based on the results of parametric studies for the evaluation of various options, the following guidelines for improved core design were made: change the structural material of the cladding and the wrapper from Ni-base alloy to stainless steel thus reducing the average fuel enrichment by 0.9% (however, possibly annealed by the increased thickness required by the lower mechanical strength); change the moderation by using zirconium hydrides (or YH_2) instead (or in complement) of water rods and increase the water gap between the sub-assemblies and filling it with cold water. Based on the assessment performed the following guidelines for an improved core were made: keep the downward water flow-since this helps flatten the axial power shape and it helps compensate the large reactivity swing caused by under-moderation and the large volume of absorbing material; increase the moderation ratio in order to reduce enrichment and the reactivity swing; finalize the mechanical design and flow paths of the fuel assembly; reduce neutron absorption by structural materials; make the sub-channel flow more uniform.

With respect to some of the design constraint criteria, it was suggested to use a nominal cladding temperature of 620 ºC, a maximum cladding temperature of 1200 ºC for class 3 and 4 transients, a maximum fuel temperature of 1930 ºC and a value of linear heat rate of 390 W/cm. More accurate design constraints values could only be determined by performing a thorough transient and safety analysis of the HPLWR. After performing the analysis of the "reference design" during the HPLWR project, it is believed that with appropriate design changes, the HPLWR core can be improved substantially by addressing the above mentioned issues.

With various and appropriate design changes which have been studied during the course of this HPLWR project, the HPLWR "reference concept" could be improved but this, however, would require a substantial design effort with all constraints taken into account. Rather complete new revolutionary designs could also be envisaged but this would even increase the overall design effort. At the conclusions of the HPLWR project, there is a much clearer understanding of what constraints apply to the design, and what is required to upgrade the necessary design tools to pursue the analyses, supported by experiments. It is envisioned that such tasks could be supported by an R&D project within the Sixth Framework Programme of the European Union. As Super Critical Water Reactor has also been identified as one of the six most promising reactors by the Generation IV International Forum (GIF), there is some chance that some R&D work could be shared in a broader context.

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