THERMAL HYDRAULIC TRADEOFFS IN THE DESIGN OF IRIS PRIMARY CIRCUIT

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ABSTRACT

IRIS (International Reactor Innovative and Secure) is currently being developed by an international consortium, led by Westinghouse and including universities (Univ. California at Berkeley, MIT, Polytechnic of Milan, Tokyo Institute of Technology, Univ. of Pisa), industries (MHI, BNFL, Bechtel, Ansaldo) and utilities (JAPC). The nucleus of the effort was provided by the US DOE Nuclear Energy Research Initiative program that funds the U.S. participants. Main characteristics of IRIS are proliferation resistance achieved through employment of a long life core; enhanced reactor safety achieved by both reducing accident initiating events and by simplified passive safety systems; while maintaining economic attractiveness.

In order to achieve high level of safety, reduce complexity and capital cost, and enhance proliferation resistance, an integral primary circuit configuration has been selected. The integral configuration (the core, steam generators, coolant pumps, pressurizer and control rods are all contained within the reactor vessel) has no loop piping and thereby eliminates the possibility of large loss of coolant accidents. If the reactor vessel and components are designed for a very high level of natural circulation, which is promoted by an integral design, the consequence of loss of flow accidents can be significantly reduced or even completely eliminated.

Core and integral primary circuit design optimization has been performed using the OSCAR computer code, a specialized tool for the analyses of the IRIS primary system developed at POLIMI.

Results of trade-off studies of various in-vessel configurations explored to achieve tight packaging and high serviceability and/or replacement of components such as steam generators and pumps are reported. Effects of changes in secondary side parameters and steam generator design on system efficiency were explored together with the optimization of the vessel and steam generator dimensions and costs. The

aim of the trade-off analyses was to limit the design space, and select a reference configuration for the IRIS reactor.

SYSTEM DESCRIPTION AND DESIGN REQUIREMENTS

In this study, design parameters of the primary circuit were varied following progression of the neutronics design and were affected by the components selection, e.g. SGs.

Core

To achieve the 8+ years core for the IRIS reactor, three different solutions were considered [1]. As the reference case for the IRIS reactor a moderated core has been considered, with a triangular pitch over diameter ratio of up to 2.0. As a future option, to increase both discharge burn-up and the power volumetric ratio of the core, a hexagonal core with a tight lattice has been considered, with two different fuel rod P/D ratios of 1.1 and 1.05 [1]. All three cases have been considered in the tradeoff studies. Standard PWR pins (8.2 mm fuel pellet diameter, 9.5 mm external cladding diameter) are assumed at this time for all three core options.

For the tight core, hexagonal assemblies were selected and for the loose lattice core both square and hexagonal assemblies are considered. There are 169 rods with 17 non-active rods per hexagonal assembly, which corresponds to around 10% of the total lattice positions occupied by control and instrumentation rods as suggested by the neutronic analyses group [1]. Open assemblies are considered.

Steam Generators

The SG design is still under development and various solutions are being considered. Following the preliminary investigation by the SGs design group a once through, cross-flow SG has been selected as the reference configuration for these analyses.

It has a triangular pitch with P/D of 1.4, an external tube diameter of 11.2 mm and 65% of the total annular area is assumed to be available for SG tubes with 6 SGs modules. The uncertainty on these values is still relatively high and the following table illustrates the range of values considered by the SG design group:

P/D with triangular pitch	1.4 to 1.51
Tube external diameter	8 mm to 12 mm
Fill Factor	45% to 80%
Number of SG modules	6 to 12

The P/D a value of 1.4 appears to be achievable as the SGs design proceeds. External tube diameter equal to 11.2 mm external tube diameter is a conservative estimate and the possibility to use a smaller diameter is currently investigated. The most serious issue is the fill factor (representing the fraction of the total annular area effectively available for the SGs tubes): a 65% fill factor appears to be achievable.

Secondary side

A typical 50 °C sub-cooling at SG inlet and a 25 °C superheating at the outlet are considered for the secondary side. Although 25 °C of superheat is considered to result in sufficiently dry vapor for once-through SGs, operational and control analyses will be needed to verify if higher outlet superheating is needed to maintain dry vapor during transients.

No secondary side pressure below that of state of the art plants is considered even for the full natural circulation case (AP600, secondary side pressure around 6.0 MPa). For forced circulation cases, the possibility of increasing the core temperatures is considered to achieve higher efficiencies. This should be made possible by several differences between IRIS and typical PWRs (i.e. lower linear power, and secondary flow inside SGs tubes, which places the tube wall in compression and limits stress corrosion problems).

Core-to-SG distance

One of the critical parameters of the thermal-hydraulic analysis is the SG-core distance. To avoid irradiation and activation of the SGs, and pumps located below the SGs, the distance between the upper part of the active core (not including the gas plenum) and the lower part of the SGs or pumps was set to be at least 1 m.

This assumption has been confirmed by preliminary POLIMI calculations on SGs activation. Given this minimum value, the actual SG-core distance is determined based on the desired level of natural circulation.

The overall objective of the thermal-hydraulic trade-off studies of IRIS was to evaluate different design solutions from the point of view of achieving the target cost performance while satisfying safety, proliferation resistance and economy. In search for such a solution, the design space was limited by the following constraints:

- **Reactor Pressure Vessel Height:** since the total weight and dimensions of the vessel are of significant importance, it is desirable to minimize them. For a 300 MWt reactor, the limiting vessel height of 17-18 m has been considered with a vessel diameter below 4 m (corresponding to a vessel weight of 300-330 ton).
- **Thermal Power:** Following preliminary economic evaluations of the system, a 150MWt system has been considered to be uneconomical, and only sizes between 300 and 900 MWt are currently considered. For the reference case, a 300MWt reactor has been selected.
- **System efficiency:** efficiency of the state of the art plants (at least 32%) was set as the minimum acceptable for IRIS. This leads to a minimum acceptable value for the secondary side pressure of around 6.0 MPa and this sets a lower limit for the inlet core temperature of ~275 °C.
- **Average Linear Power:** based on the preliminary neutronics design, a maximum average linear power (for fuel pellets of 8.2 mm in diameter) of 12-13 kW/m was selected.
- **Safety:** from the safety point of view, the objective of the thermal-hydraulic design is to eliminate LOFA events as Class IV accident initiating event and

minimize the potential for or the consequences of other Class IV events. To achieve this, an appropriate degree of natural circulation would be required.

- **Development effort:** to enable fast deployment of the IRIS, a preference is given to proven design solutions or those requiring minimum development effort. Higher potential, higher risk technical solutions are reserved as potential upgrade options.

THE OSCAR CODE

The OSCAR 2.0 (Optimization Simplified Code for the Analysis of Integral Reactor) code [2] employed here for the IRIS analyses was developed at the Polytechnic of Milan. It is a simple, yet powerful tool for the preliminary investigation of integral type reactors and of the NILUS [3] and IRIS reactors in particular.

Version 2.0 of the code was completed on July 2000: and included new computational models for the steam generators and more detailed CHF calculations and included a new graphic user interface. The OSCAR code uses a one-dimensional thermal-hydraulic model to evaluate the main dimensions of an integral reactor: The vessel is divided into different zones (core, riser, coolant pumps, SGs, downcomer) with various engineering options left open for the user selection (i.e. type of SGs, core configuration, etc). Figure 1 shows the main structure of the code and the main zones of the primary circuit.

The code solves steady-state mass, energy and pressure balance equations for the primary circuit components, with detailed models developed for the SG and DNB analysis, and with a complete library of possible core layouts (both hexagonal and square fuel assemblies).

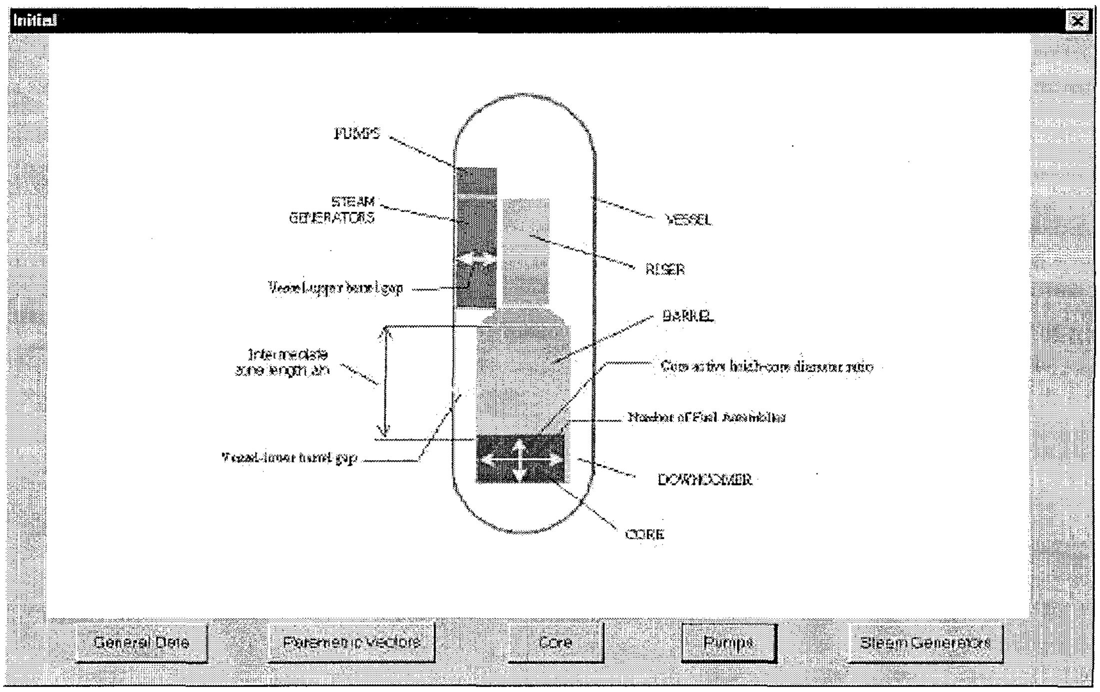

Figure 1: OSCAR code main screen

The IRIS system analysis is then performed in the following steps:

Core dimensioning

The core dimensioning utilizes a library of pre-calculated assembly layouts, for both hexagonal and square fuel elements to determine the total size of the core, the internal barrel diameter, and the total reactor height from the user defined core height/diameter (H/D) ratio (based on neutronics analyses [1]).

The user also defines the core geometry (triangular or square lattice, number of elements, number of fuel pins per element side, number non-active pins per element), total power, inlet and outlet temperatures. The code then evaluates the mass flow rate, the average linear power, the core active height, the core diameter and other relevant core dimensions. Finally, a hot channel analysis for CHF calculation is performed.

SG evaluation

A main assumption in IRIS is to consider modular, once-through SGs. The SG modules are dimensioned via a refined model: a detailed description of the models and procedures adopted in the OSCAR code is given in [2].

The user can select different parameters for the SG modules, related to number of modules, header geometry, tube bundle type, flow type in the shell side. Once the vessel geometry, primary and secondary flow parameters have been selected the code performs the SG dimensioning and thermal-hydraulic performance calculations.

The numerical solution adopts a fine grid model. Heat transfer analysis is solved via a library of correlations for the secondary side [2] from subcooled boiling to post-crisis heat transfer. State-of-the-art correlations and empirical models are adopted for the primary side.

Integral Reactor Vessel dimensioning

The dimensioning of the reactor vessel is performed by considering all the reactor vessel zones (core and core grids and plates, riser, SGs, downcomer, and pumps if present). Since the core, SGs, and corresponding downcomer and riser zones are already dimensioned due to thermal-hydraulic and design constraints, the resulting core-to-SG gap distance is evaluated in order to satisfy the whole momentum balance for the primary flow path. In full natural circulation conditions, this gap is necessary to establish the distance between thermal centers to achieve the pressure head needed to sustain the reactor flow rate. In case of aided forced circulation, the supplementary pump head, which is necessary to close the momentum balance, is calculated.

For the SGs and the core, detailed models and correlations are used for pressure loss evaluations.

Once the thermal-hydraulic evaluation is completed, the OSCAR code evaluates the reactor vessel and SG header thicknesses and weights and total water and steam volumes.

Efficiency Evaluation

The Oscar code uses a simplified approach to plant efficiency evaluation by calculating the efficiency as 75% of the efficiency of the ideal Carnot cycle between SG outlet temperature and the condensation temperature.

While this approach is not appropriate for design purposes, it has been used for its simplicity. A more detailed approach will be developed that takes into account the steam extraction points on the turbine for condensate reheat and the heater drain flows, thus allowing more suitable investigations into the BOP will be employed in future analyses.

Results from this simplified correlation have been compared with data from a number of plants (e.g. AP600 design, Sequoyah plant) and appear to be in a good agreement (within an average difference of 0.2% percentage point in efficiency values).

The correlation has been improved considering Westinghouse plant efficiency data to take into account the steam superheating of the IRIS once-through SGs. The resulting correlation used in the OSCAR code is thus:

$$\eta = 0.75 \cdot \eta_{Carnot}\left(T_{sat,SG}, T_{cond}\right) + 0.025 \cdot \left(\frac{T_{out,SG} - T_{sat,SG}}{100}\right)$$

NATURAL CIRCULATION AND FORCED CONVECTION

The primary coolant circulation mode for the reactor has been investigated using the OSCAR code for the three different core configuration proposed. To maximize the effect of natural circulation a core inlet temperature of 275 °C and a core outlet temperature of 330 °C were considered. The possibility of achieving a Full Natural Circulation (FNC) mode in the primary flow circuit has been investigated. The results of the analysis are summarized in Figure 2 and Table 1. For the moderated thermal core, a P/D of 1.45 has been considered: this corresponds to a moderation ratio typical of a standard PWR.

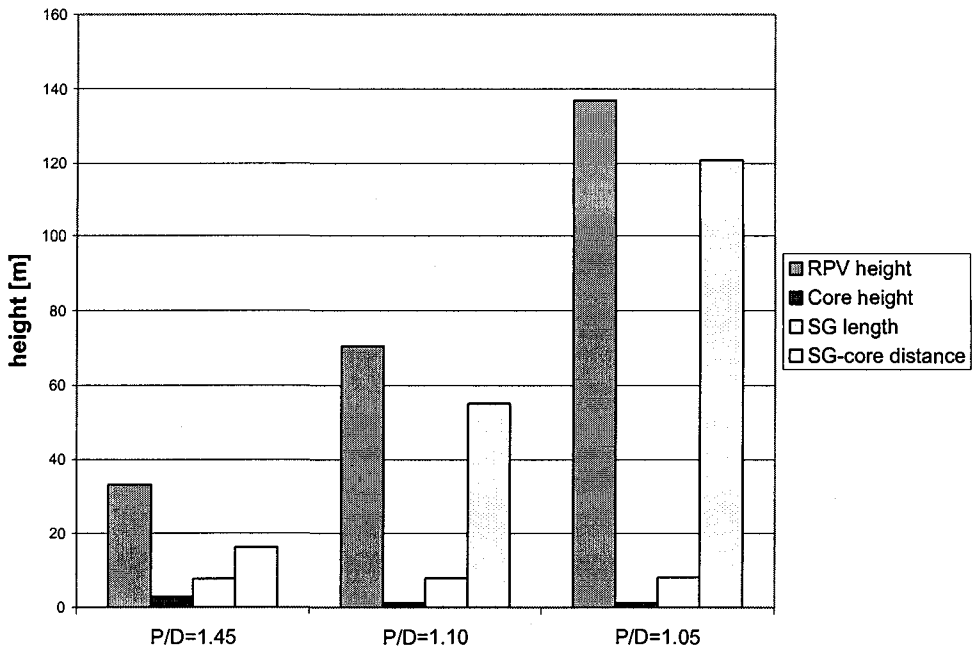

Figure 2: Reactor Pressure Vessel dimensions for IRIS-300MWt core for different P/D ratios.

	P/D=1.45	P/D=1.10	P/D=1.05
Sec. Side Pressure	6.0 MPa	6.0 MPa	6.0 MPa
NCR	100%	100%	100%
Average Linear Power	10.56 kW/m	10.05 kW/m	10.53 kW/m
Vessel Diameter	3.98 m	3.82 m	3.72 m
Vessel Weight	618.25 ton	1260 ton	2355 ton
DeltaT core	55 C	55 C	55 C
SG pressure losses	17.8 kPa	20.2 kPa	21.9 kPa
Core pressure losses	4.3 kPa	38.9 kPa	101.2 kPa

Table 1: main data for the IRIS-300MWt core for different P/D ratios

For tight lattice cores, the FNC solution is completely ruled out. For the moderated thermal core however the results of the analysis are not definitive. It is clear that achieving FNC primary flow is not possible considering the design limits set for the analysis. However, to make this option feasible, different options are available: increasing the annular area to reduce SGs pressure losses, decreasing the linear power to reduce the core flow rate and pressure losses, even though their effect is not significant given the loose lattice and low mass flux, and a low pressure losses SG design. Obviously all these deviations from the design limits would lead to an increase in cost, and it is questionable whether this increase would be comparable to the savings in capital and maintenance costs due to elimination of the coolant pumps. Moreover the possibility of higher power IRIS would make achieving FNC even more costly. Results of preliminary parametric analysis suggest that FNC would be a viable solution for very low power (about 150MWt), but with unclear advantages for higher power options (between 150 and 300-400 MWt), and a clear economical disadvantage when higher powers are considered (from 300/400 to 900 MWt).

In the case of a tight lattice core, FNC is not achievable even at very low power, unless very low linear powers and extremely flat (low H/D ratio) cores are considered.

Moreover, the very high core ΔT and the difficulty to orifice individual assemblies for the very long core life introduce a high level of uncertainty associated with achieving such a high core outlet temperature. For all the cases considered, a positive outlet quality of 4.4% is maintained for the hot channel. While a detailed sub-channel analysis is needed to evaluate the Minimum Departure from Nucleate Boiling Ratio (MDNBR), preliminary analysis considering several different DNB correlations suggest that this outlet temperature would not be acceptable.

The possibility of more extensive boiling to achieve FNC has been investigated. Unacceptable MDNBRs evaluated together with several other technological problems associated with boiling suggested that only very low local degree of boiling could possibly be acceptable.

For these reasons, FNC has been discarded and a standard forced circulation option is selected. Given this selection, increased mass flow rate and reducing the core ΔT maximizes the plant efficiency, improves the MDNBRs, and prevents even local boiling.

Even with forced circulation in the primary flow circuit, the degree of Natural Circulation Ratio (NCR) is important: The NCR is defined as the ratio of the total net buoyancy head over the total pressure losses in the circuit at nominal conditions.

All the considered configurations have NCR much larger than those typical of loop PWR, and thus the term Aided Natural Circulation (ANC) has been introduced. The effective degree of NC will be determined following safety analysis of the system, so to mitigate or eliminate various possible accident scenarios: in IRIS NC is a safety feature, which will be explored and evaluated so to maximize its effects.

PUMPS SELECTION AND LAYOUT

The selection of forced circulation requires careful consideration of two different issues: a) whether and how can LOFAs be eliminated as Class IV events (limiting faults, i.e. accidents of low probability that may lead to fuel failures) and b) where to position the pumps in the RPV.

The most serious LOFAs accidents result from a postulated Locked Rotor or Shaft Seizure (LRSS) events, which lead to an almost instantaneous failure of one coolant pump and the resulting sudden reduction in core flow rate. To eliminate these two accidents the most intuitive solution is to eliminate the pumps by designing a FNC reactor. However the same positive effect can be achieved by increasing the NCR: if for example there are four pumps in the system and a NCR of 20%, NC would actually be equivalent to a supplementary virtual pump. Thus, the loss of one pump following a LRSS event would be equivalent to the loss of 1-out-of-5 pumps rather than 1-out-of-4 in case of a system with no NC.

This suggest another possibility to limit the possibility of a class IV LOFA event: in fact the same effect can be achieved by increasing the number of pumps, and for IRIS this would have a lower cost that in a loop PWR since there is no need for supplementary piping.

Finally, another way to reduce the consequences of a LRSS is by having pumps with a flat characteristic so that following the loss of a single pump the total flow rate would not change significantly. This is possible in IRIS because of the low pump head requirement as discussed earlier. Coupling the number of pumps, the high NCR and the characteristic of the pumps, the LRSS event will be eliminated as a Class IV event for IRIS.

The second main issue associated with pumps is their location. Locating the pumps above the SGs would eliminate low penetrations in the RPV and greatly simplify O&M for the pumps. However the fluid will be pumped at hot conditions and therefore with a higher risk of cavitation. Given the very low pressure head required, and thus the limited net positive suction head requirements, and the requirement of avoiding low vessel penetrations, an options with the pumps above the SGs(as in the

SIR reactor [4]) is tentatively selected. This choice should also lead to simpler inspections and maintenance operations on the pumps.

CORE INLET AND OUTLET TEMPERATURES

Given the selection of forced coolant circulation with ANC, the possibility of increasing the core temperatures so to maximize efficiency has been evaluated, especially considering that maximum core temperature for loop PWR is essentially limited by stress corrosion concerns for the SG tubes. This is of a less concern in IRIS since the primary (high-pressure) fluid is on the outside of the tubes, so that the tubes are in compression. Therefore, a core outlet temperature of 330 °C has been considered for the reference case.

Two opposite effects need to be considered in choosing the core inlet temperature. On one hand, increasing the core inlet temperature decreases the NCR of the primary circuit, since it decreases the buoyancy head and since it leads to an increase in mass flow rate and pressure drops. On the other hand, increasing the core inlet temperature allows for an increase in SG steam pressure and efficiency, or allows a reduction in SGs heat transfer surface. Also, an increase in the core mass flow rate increases the MDNBRs for the system.

Given the requirement of reasonably high NCR, a core inlet temperature of 292 °C has been selected for the reference configuration. More detailed analysis will be developed using the VIPRE subchannel code to evaluate MDNBRs for the system and to optimize the core temperatures.

INCREASING THE NATURAL CIRCULATION RATIO

The possibility of increasing the NCR has been considered, should such a requirement be suggested by the safety analyses. Essentially two different ways are available for increasing the NCR of the system: either increasing the height of the RPV or decreasing the core linear power (thus reducing the core pressure losses, but having the drawback of increasing the diameter of the core, and thus the diameter and weight of the RPV).

Figure 3 shows the cost of each of the different solutions for the tight lattice core, assuming the cost to be proportional to the weight of the RPV. The Locus of minimum heights represents the weight of the minimum size vessel (as optimized by the OSCAR code) for different average linear powers. The dotted lines represent the effect on the NCR of increasing the vessel height while keeping the same linear power.
In this case, it can be clearly concluded that the more economical way to increase the NCR of the system is by reducing the core linear power. For IRIS this could also have the advantage of maximizing the core life.

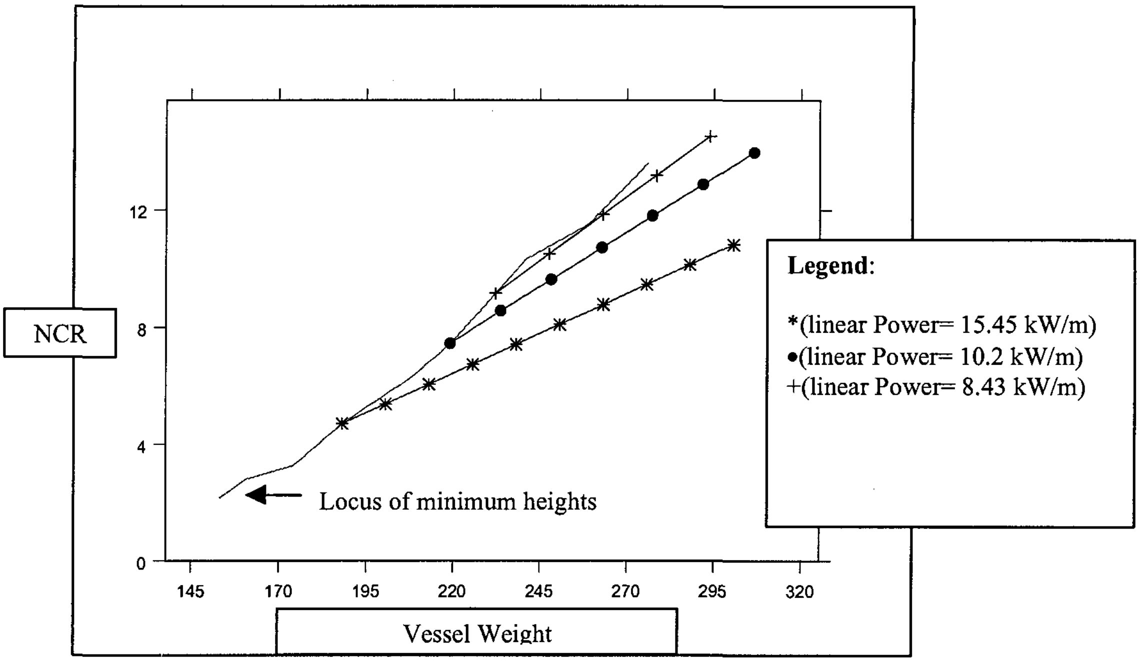

Figure 3: NCR as a function of vessel height and average linear power. P/D=1.1

The same conclusion is not true for the thermal. In this case the system is not ruled by the pressure losses in the core, but rather by the pressure losses in the SGs so that reducing the linear power does not have a significant effect on the NCR. Therefore, it has been concluded that for the reference configuration the maximum average linear power compatible with neutronic requirements (i.e. between 10 and 12 kW/m) will be considered, and in the event that a higher degree of NC is required by safety analyses, an increase in the RPV diameter to increase the annular area thus reducing the SG pressure losses will be evaluated.

CONCLUSIONS

Following these preliminary trade-off analyses, the reference layout for IRIS RPV is shown in Figure 4. The reference thermal hydraulic design parameters are summarized in Table 2.

Considering the selected RPV layout and reference configuration, detailed thermal-hydraulic analyses will be performed to establish the optimum point design and the thermal-hydraulic characteristics of IRIS.

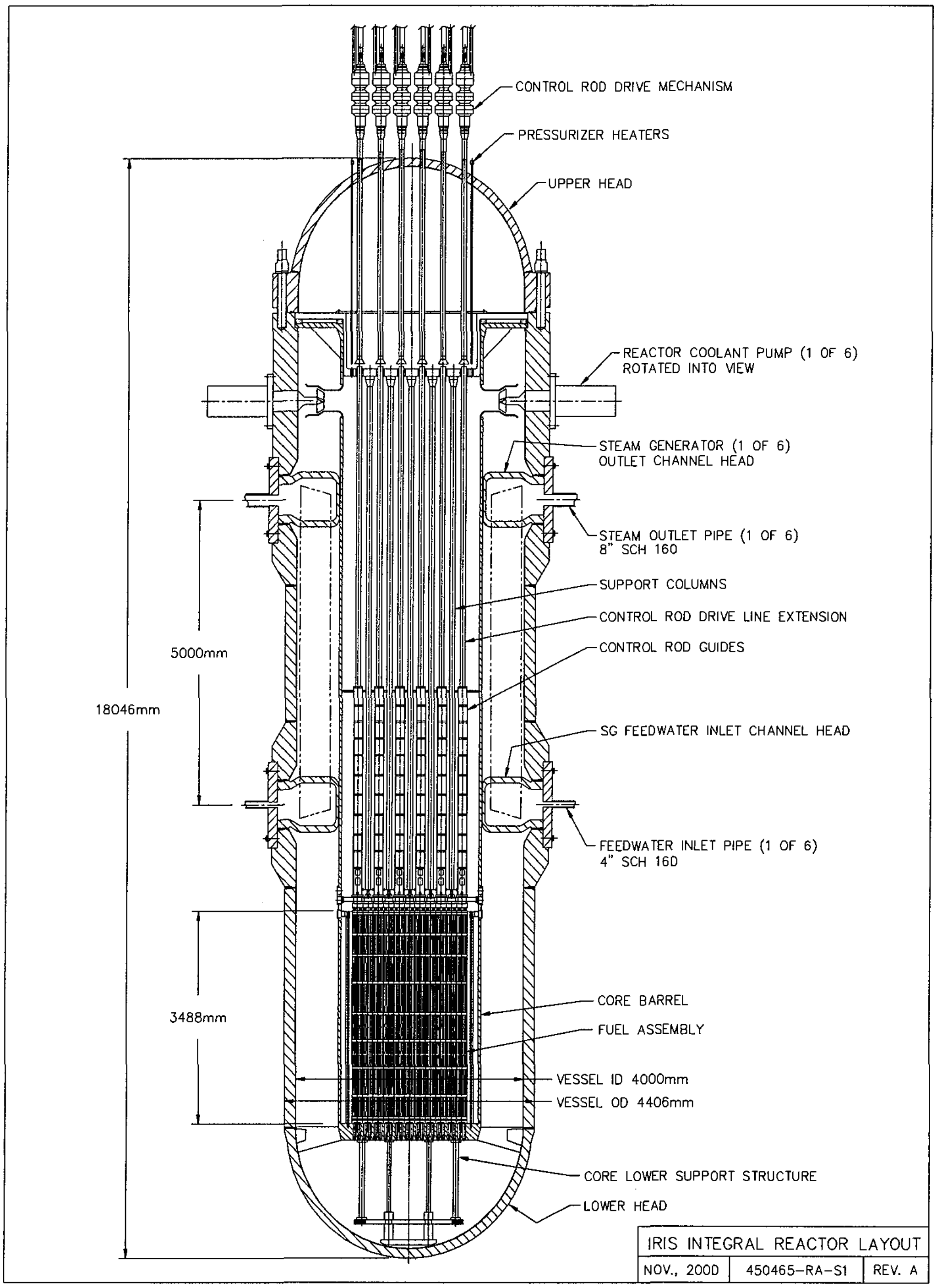

Figure 4: IRIS reactor reference layout

Table 2: Main IRIS Thermal-Hydraulic Data.

Main characteristics	
Thermal Power [MW]	300
Efficiency	34.6%
Average Linear Power [kW/m]	12.4
Natural Circulation Ratio $(\Delta\rho\ g\ h)/\Delta p$	26.1%
Primary Pressure [MPa]	15.5
Secondary Pressure [MPa]	7.0
Number of SGs	6
Number of Pumps	6
General Thermal hydraulic characteristics	
Primary side mass flow rate [kg/s]	1345
Secondary side mass flow rate [kg/s]	160.9
Inlet core temperature [C]	292
Outlet core temperature [C]	330
SG inlet temperature [C]	236
SG saturation temperature [C]	285.8
SG superheating [C]	25
Condensation temperature [C]	32.6
Core pressure losses [kPa]	3.8
SGs pressure losses [kPa]	18.1
Total pressure losses [kPa]	26.0
Steam Generator Data	
Lattice type	Triangular
P/D	1.4
Total number of tubes	22000
Tube external diameter [m]	0.01120
Tube material	Inconel TT690
Core geometry and main characteristics	
Core active height [m]	3.0
Total core height [m] including the fission gas plenum	3.5
Lattice type	Hexagonal
P/D	2.0
Vessel data	
Total Vessel height [m] (excluding thickness)	18.1
Vessel internal diameter [m]	4.06
Internal Barrel diameter [m]	2.42

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