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On applicability of plate and shell heat exchangers for steam generation in naval PWR

Luciano Ondir Freire*, Delvonei Alves de Andrade

Instituto de Pesquisas Energéticas e Nucleares (IPEN-CNEN/SP), Av. Professor Lineu Prestes 2242, 05508-000 São Paulo, SP, Brazil

HIGHLIGHTS

- Given emissions restrictions, nuclear propulsion may be an alternative.
- Plate and shell heat exchangers (PSHE) are a mature technology on market.
- PSHE are compact and could be used as steam generators.
- Preliminary calculations to obtain a PWR for a large container ship are performed.
- Results suggest PSHE improve overall compactness and cost.

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ABSTRACT

The pressure on reduction of gas emissions is going to raise the price of fossil fuels and an alternative to fossil fuels is nuclear energy. Naval reactors have some differences from stationary PWR because they have limitations on volume and weight, requiring compact solutions. On the other hand, a source of problems in naval reactors across history is the steam generation function. In order to reduce nuclear containment footprint, it is desirable to employ integral designs, which, however, poses complications and design constraints for recirculation type steam generators, being interesting to employ once through steam generators, whose historic at Babcock & Wilcox is better than recirculation steam generators. Plate and shell heat exchangers are a mature technology made available by many suppliers which allows heat exchange at high temperature and pressure. This work investigates the feasibility of the use of an array of welded plate heat exchangers of a material approved by ASME for pressure barrier (Ti-3Al-2.5V) in a hypothetical naval reactor. It was found it is feasible from thermal-hydraulic point of view and presents advantages over other steam generator designs.

1. Introduction

Shipping was responsible for 2.7% of global CO_2 emissions in 2007 (Gravina, 2012), and it is responsible for 4–9% of global emissions of SO_x and for 15% of global emissions of NO_x (Eyring, 2010). Because of that, countries now start to enforce more restrictive regulations which should lead to rise of fuel costs, which along the depletion of fossil fuels, has a huge impact on shipping rates and profitability. Furthermore, the prices for fossil fuels fluctuate, adding uncertainties for the shipping market.

Taking into account that shipping is responsible by 95% of global commerce (Royal Academy of Engineering, 2013), it substantially impacts on economy and prosperity.

Nuclear energy is free of green house emissions; in the opinion of many experts, including Jenkins (2011), is the only emissions free energy which can replace fossil fuel with current technologies. Nuclear fuel is currently cheaper and has relatively stable prices compared to fossil fuels. Another side advantage is the reduction of frequency and gravity of oil spill at sea.

Nuclear power is a proven technology for naval propulsion, with a history of about 700 nuclear naval reactors worldwide (Royal Academy of Engineering, 2013) for military use, differently from other clean technologies that are not still proven. Currently, about 200 nuclear naval reactors are in operation (Gravina, 2012). Besides, naval reactors have higher operational reliability when compared to diesel motors (Carlton, 2011) and allow better flexibility in

* Corresponding author. Tel.: +55 11 4063 4516.
E-mail addresses: luciano.ondir@gmail.com (L.O. Freire), delvonei@ipen.br (D.A.d. Andrade).

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design and operation, given the smaller volume and less frequent refueling.

Pressurized water reactors have proven to be simple, safe and compact, allowing more space for goods deadweight. Generation IV plants are simpler, safer and more economic, being the estimated costs nearly halved according to Westinghouse. Special attention for naval applications should be paid to integral reactor types, which have less possible accidents, smaller shielding size and weight. A great advantage of naval reactors over land nuclear power plants is the presence of a reliable heat sink – the sea.

2. Assumptions

This section lists a series of assumptions made in order to make this work applicable from the point of view of nuclear power plant vendors and merchant ships builders and operators.

2.1. Naval reactors requirements

Nuclear power has high capital costs. Therefore, it should be more interesting on classes of ships which demand high propulsion power. Making a fast market research, it was found that a numerous class of merchant ships with high propulsion power is the Post-Panamax class, which requires a propulsion power of about 60 MW (MAN Diesel A/S, 2008). This class was responsible, in 2008, by 32% of transportation of goods (MAN Diesel A/S, 2008). Adopting a conservative efficiency of 25% – Angra 2 from Brazil claims 35.8% of efficiency (Eletronuclear, 2014) – for the complete cycle, it is required a thermal power of 240 MW from the reactor.

In order to reduce volume and weight, the reactor should be integral type. It must also be possible to realize periodic in service inspections.

2.2. Avoidance of new technologies

This work aims at proposing a feasible solution within the present technologies, changing only integration aspects. New technologies need to be avoided as they represent risks. Therefore, only mature and well known technologies must be chosen to reduce risks. The only successful naval reactor technology up to now is the pressurized water type, due its simplicity, compactness and safety.

2.3. Core configuration

It is intended to employ the proven 17 × 17 square lattice UO_2 fuel elements with Zircalloy cladding. The coolant must not have soluble boron during normal operation because it would require great volumes. The reactivity is controlled by burnable poison and control rods.

The primary working pressure and temperature are assumed the same as Angra 2 nuclear power plant (157 bar and 308.6 °C) (Eletronuclear, 2014). Additionally, the following conservative assumptions are considered:

- There is not intermixing between channels. This assumption greatly simplifies analysis and does not affect the results given the required precision for this study. According to Reddy and Fighetti (1983), ignoring intermixing leads to under prediction of critical heat flux and increased the standard deviation of the prediction for a given subset from 12% to 14%. Therefore, for a feasibility study, this level of imprecision is acceptable.
- The radial peak factor is 1.7, the axial peak factor is 1.6.
- The power distribution in the hot channel is a sine without axial offset, as seen in Fig. 1, which is a conservative assumption from the point of view of critical heat flux.

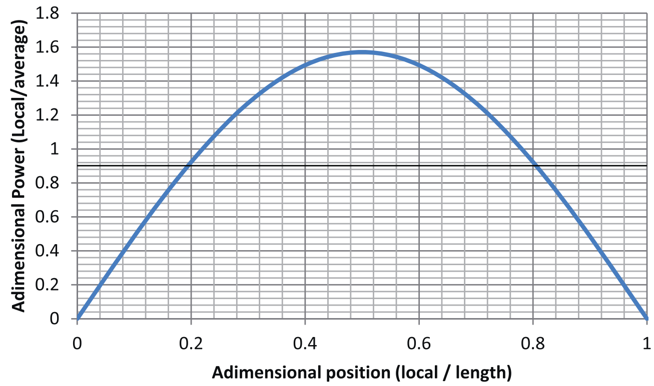

Fig. 1. Assumed axial power distribution on hot channel.

- The fuel elements have eight spacer grids, each one with a charge loss coefficient of 2, the upper value of the range of typical values (Reddy and Fighetti, 1983).

To choose the relation height over radius K, it was adopted the option of optimizing the buckling, which is, for a bare cylinder (Ippolito, 1987):

$$B^2 = \left(\frac{2.405}{r}\right)^2 + \left(\frac{\pi}{h}\right)^2 \quad (1)$$

Expressing a given core volume as a function of h and K:

$$V = \pi \frac{h^3}{K} \quad (2)$$

Expressing a given core volume as a function of r and K:

$$V = \pi r^3 K \quad (3)$$

Assuming the volume is unitary, rearranging Eqs. (2) and (3) and substituting in Eq. (1):

$$B^2(K) = 2.405^2 \pi^{2/3} K^{2/3} + \pi^{8/3} K^{-4/3} \quad (4)$$

The derivative of $B^2(K)$ is:

$$\frac{dB^2(K)}{dK} = 2.405^2 \pi^{2/3} \frac{2}{3} K^{-1/3} - \frac{4}{3} \pi^{8/3} K^{-7/3} \quad (5)$$

The optimum (minimum) buckling is at null derivative, which holds the relation:

$$K = \frac{h}{r} = \sqrt{2} \frac{\pi}{2.405} \cong 1.85 \quad (6)$$

This relation, obtained for a bare cylinder, is assumed as representative of the core for preliminary sizing purposes. Ippolito (1987) adopted a value of 2.5 for K, as the core of NS Savannah. However, for naval propulsion, it should be adopted a core as flat as possible for integration purposes. As the buckling function is quite flat near the optimum, as shown in Fig. 2, it should be interesting to pick the lowest K ratio. The buckling for a core with $K = 2.5$, which was employed at NS Savannah (Ippolito, 1987), is about 3.85% bigger than the optimum value. The same level of buckling degradation on the lower side brings a value of 1.39 for K, which was adopted for this study due to integration issues.

2.4. Steam generators configuration

It is adopted a configuration of an array of rectangular plate heat exchangers arranged in the annulus of the vessel, with the faces toward the center. The primary and secondary fluids are in counter current as in once-through steam generators, which have better availability than recirculation steam generators (Wade, 1995). The

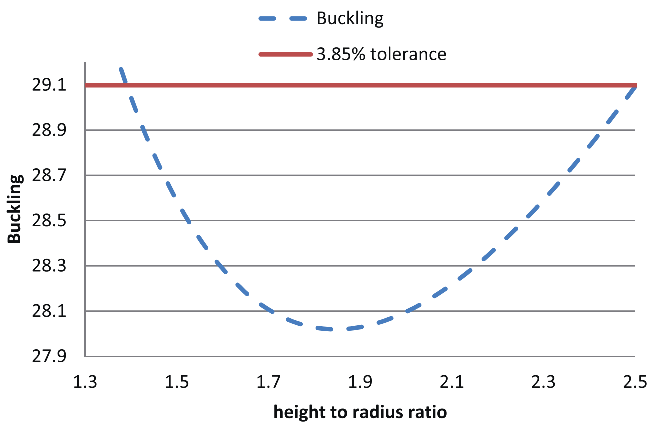

Fig. 2. Buckling as a function of height to radius ratio.

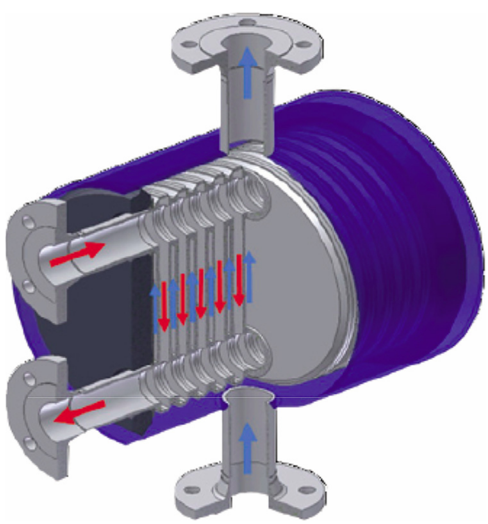

Fig. 3. Plate and shell heat exchanger.

feed water is distributed in the lower part of the plate heat exchangers and the steam is collected in the upper part. The plates are 1 mm thick, because it was the maximum plate thickness found in some suppliers, and made of Ti-3Al-2.5 V, because this is an alloy accepted by ASME code for pressure barrier of pressurized water reactors and its properties are provided by the standard, even though its thermal conductivity is not the best amongst the titanium alloys. The plates are assumed as formed and welded together, like in a plate and shell heat exchanger shown in Fig. 3 (ViFlow, 2014). A number of suppliers state to be able to provide plate and shell heat exchangers for temperatures beyond 400 °C and 100 bar of design pressure, as seen in Table 1, so it is assumed to be feasible to adopt a baseline of maximum pressure difference between primary and secondary side of 100 bar (the reactor protection system shuts down the reactor in case of subcooling greater than 100 bar) and maximum temperature of 350 °C. The secondary pressure is assumed as 64.5 bar. The maximum steam speed in the pipes is 200 m/s and the maximum water speed is assumed as 3 m/s to avoid noise, erosion and charge losses. It is assumed the feed water is saturated, which enhances the heat flow.

The primary coolant channels have 3 mm thickness and the secondary fluid channels have 1 mm thickness. It is adopted a margin of 42% of the surface for structural support. As Yoon et al. (2014) state the straight channel is the less efficient type for thermal transfer and may be the best option in applications where the charge losses must be as low as possible, it is assumed the plates have only straight channels to be conservative, which in turn reduces the analysis effort.

It is also assumed the heat exchangers are rectangular blocks and they need a 5 cm space between them to allow fixation by screws. At opposing sides, two safety steam generators of shell and tube technology do not allow the arrangement of the plate and shell exchangers in all the annulus.

The roughness in the plates is assumed as 45.7 μm, because it is quite a standard value for steel and titanium plates.

In order to compensate for fouling, a margin of 25% on the required heat transfer area is adopted for the steam generator.

2.5. Steam generators safety function

Typically in stationary PWR, the steam generators play a dual role: they generate steam to be used on power conversion system and in case of design basis accidents, they play the emergency secondary heat removal function. This solution imposes constraints in design and operation of steam generators, like the number of two or more steam generators, periodic inspections, material classification, segregation, redundancy and independency requirements. If the safety function of emergency secondary heat removal is played by other equipment, not used during normal operation (adoption of functional segregation), the steam generator used for power conversion does not need to be redundant nor classified, provided its failure do not impair the safety functions. The typical steam generators cause loss of coolant accidents in case of failure of their tubes, so there is not great advantage in functional segregation because they need to be classified anyway. But for plate heat exchangers, depending on how the flow is organized (the plates working in compression by primary fluid, which precludes cracking propagation), the catastrophic failure becomes virtually impossible and only small leakages may happen with a frequency above that allowed for design basis accidents. In this case, the association of a steam generator for power conversion and other two small steam generators used only for design basis accidents scenarios may be safer than current practice, once those safety steam generators will not suffer corrosion wear during service life.

3. Method

3.1. Core analysis models

The most important aspect on thermal-hydraulic core sizing is the critical heat flux calculations. There are many well known correlations, like W3 (Tong, 1967) and EPRI correlation (Reddy and Fighetti, 1983). Weisman and Ying (1985) provide a method to be used along with local flow conditions calculated by COBRA code. It

Table 1
List of suppliers for plate and shell heat exchangers.

Supplier	Maximum temperature (°C)	Maximum pressure (bar)	Site reference
Viflow	500	150	http://www.viflow.fi/uploads/ViFlow_GESMEX_GB_HiRes.pdf (2014)
Vahterus	899	150	http://www.vahterus.com/sites/default/files/pshe_eng.pdf (2014)
Sondex	400	40	http://www.sondex.net/Files/Billeder/PDF/SAW%20SPS%20enkelt%20sidet2.pdf (2014)
Alfa Laval	538	100	http://www.alfalaval.com/solution-finder/products/alfa-disc/Documents/AlfaDisc.pdf (2014)

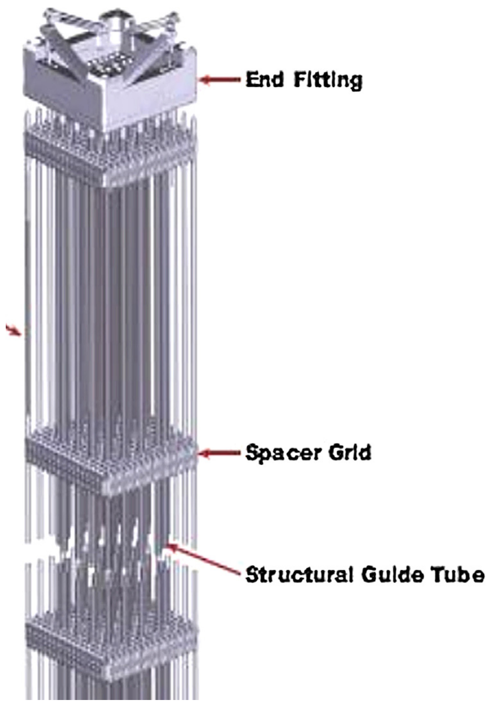

Fig. 4. Westinghouse 17 × 17 fuel assembly adopted for calculation.

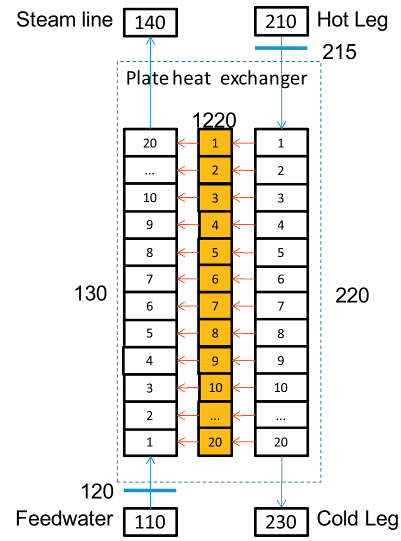

Fig. 5. Plate steam generator RELAP model.

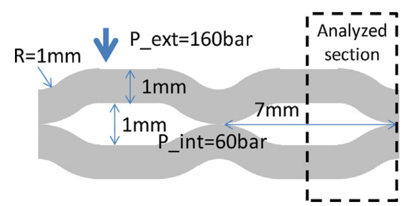

Fig. 6. Sketch of a pair of sheets.

reports a better accuracy than Tong (1967) but not better than EPRI correlation. Additionally, EPRI correlation, which also uses local conditions calculation with COBRA-3 code, is adequate to a wider range of parameters than W3. Therefore, it was chosen as the correlation to be adopted. As COBRA-3 is not readily available, it was adopted a similar model coded in Excel worksheets. Fig. 4 shows some features of the fuel assembly considered for this work.

The core head losses were calculated adding the spacers and bare rod subchannel losses. For bare rod subchannels, it was used Cheng and Todreas (1985) model for bare rod subchannel friction factor constants in square array with turbulent flow and pitch over diameter $P/D = 1.325$.

3.2. Steam generators analysis models

The steam generators are pre-sized using approximate analytical methods. The friction factor constant in the primary side is calculated using Moody diagram. The heat transfer coefficients are estimated using the Gungor and Winterton (1986) correlation and the charge losses in secondary side are estimated using Quibén et al. (2009) correlation for flattened tubes. It is believed that, for mass velocities beyond 200 kg/m^2 s, the heat transfer coefficients are similar to those obtained in circular tubes correlation, as it was found by Tibiriçá et al. (2012). In a second iteration, the heat transfer and charge losses in the heat exchanger are estimated using a simple RELAP model shown in Fig. 5.

It is also studied the models proposed by Cioncolini and Thome (2012a,b) for annular flow, which was verified to be the dominant regime for this application.

It was applied the parallel plates (ORNL, ANS reactor) model along its inter phase drag model (Scientech Inc., 1999).

It is also interesting to check the critical heat flux in the steam generator in order to know if the heat transfer coefficients correlations are applicable. Revellin et al. (2009) report that the best correlation for critical heat flux with water is the one proposed by Zhang (2006) which is adopted for this work.

It was also verified using finite elements analysis if the mechanical stresses caused by the pressure in plates are acceptable for pressure barrier components (safety class 1 according to the American ANSI/ANS standards) accordingly to ASME code. In order to reduce computational costs, it was chosen a half section of a channel, as shown in Fig. 6.

Although it is assumed the plates are welded together, the welds are made only at the inlets and outlets of feed water and steam, respectively, in order to avoid the primary coolant to mix with secondary fluid. The corrugations at the middle of the plates are just in contact with each other.

4. Development

The first step is to estimate a core able to provide the power needed. Starting from the assumptions of power needed, axial power distribution, radial peak factor, working pressure, working temperature, it was obtained a theoretical minimum core size to respect the minimum departure from critical heat flux ratio, along with a minimum core flow and temperature difference between cold and hot legs. With the core size and assumptions on spacer grids, it was estimated the core charge losses.

From reactor thermal power and water latent heat, it is estimated the steam mass flow. With the mass flow and maximum flow speed in pipes, it is estimated a minimum pipe diameter and it is chosen a commercial diameter along its schedule (wall thickness).

From the core diameter and pipes chosen diameter, it is estimated the annulus internal diameter.

An external annulus diameter is guessed for each iteration. Given the internal and external downcomer annulus diameter, it is calculated what is the best number of heat exchangers to explore efficiently the available volume, along its horizontal dimensions (width and length). As there are two safety steam generators, it is assumed the even number nearest to optimum choice.

Table 2
Parameters input to RELAP model.

RELAP model parameters	Value	Unit
Secondary side		
Mass flow	0.1366	kg/s
Flow area	0.000501038	m^2
Volume minimum length	0.025257491	m
Maximum number of volumes	87	
Chosen volume length	0.1101	m
Primary side		
Mass flow	1.637	kg/s
Flow area	0.002056891	m^2
Volume minimum length	0.051175337	m
Maximum number of volumes	43	
Chosen volume length	0.1101	m
Chosen number of volumes	20	

From the assumptions for the channels, it is calculated the number of channels, the heated perimeter, the flow area for primary and secondary fluids, the hydraulic diameters. Using also the assumed working temperature and pressures, it is calculated the flow speeds, Reynolds numbers and heat transfer coefficients. For the secondary side, in a first approximation, it is solved the local heat transfer coefficients and charge losses along the plate. Adding the heat resistance due to plate thickness and material properties, the global resistance to heat flow per area is obtained. Using the global resistance and temperature difference between the fluids, it is calculated the required heat transfer area and the heat exchangers height.

From heat exchangers flow area, hydraulic diameter, Reynolds number, roughness and length, it is calculated the charge losses. If the charge losses in the primary side of the steam generator are too great, a larger external annulus diameter is adopted and new calculations are necessary. The pumping power is estimated from the charge losses on the primary side.

Additionally, for confirmation of results, it was developed the RELAP model. For the primary side, it was adopted a gap of 3 mm and a span of 8 mm. For the secondary side it was adopted a gap of 1 mm and a span of 6.5 mm. Additionally, the time step control was set as follows:

- the hydrodynamics advancement uses a mass error analysis to control the time step between the minimum and maximum time step;
- the heat conduction/transfer time step is the same as the hydrodynamic time step;
- the heat conduction/transfer and hydrodynamics are coupled implicitly;
- the nearly-implicit scheme is used to advance the hydrodynamics;
- the test for convergence of a steady-state calculation is not made.

Table 2 provides further details about the RELAP model, which represents only one channel of 1 mm thickness of the secondary side, two plates and two half primary channels of a single heat exchanger.

It was also analyzed if the mechanical stresses caused by the pressure in plates are acceptable for pressure barrier components (safety class 1 according to The American ANSI/ANS standards). Free university student Autodesk Inventor software was employed to perform finite element analysis on a small section of the plate. According to ASME code, for the temperature of 315 °C, the allowable Von Mises stress is 140 MPa for class 1 components made of the chosen titanium alloy. It was applied an external pressure of 166 bar (the pressure at which Angra 2 opens its first relief valve) and an internal pressure of 64.5 bar (normal secondary operating pressure). It was applied symmetry boundary conditions on all other faces. The contact between the two plates was considered as a separation (just two independent pieces in touch with each other).

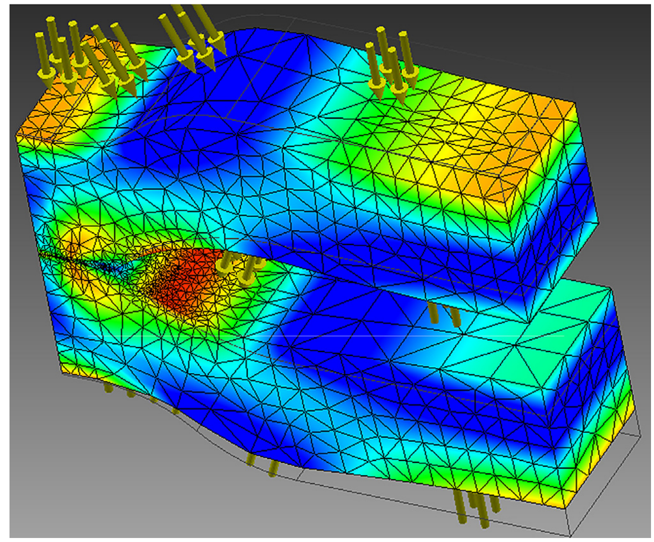

Fig. 7. Finite element analysis mesh.

The mesh adopted had a total of 61,139 nodes and 37,778 elements. The mesh is presented in Fig. 7.

5. Results

The core for the required power has the characteristics shown in Table 3.

From the core diameter, it was possible to obtain the dimensions of the annulus where the plate heat exchangers are arranged. Using the assumptions of lateral space for screws, and guesses about external annulus diameter to obtain a reasonable pressure drop on primary side, it was obtained a curve of space efficiency as a function of number of plate heat exchangers, as shown in Fig. 8.

With the optimum number of 10 heat exchangers, using the internal and external annulus radius, it was calculated the possible width and length for each heat exchanger, which along other assumptions, allowed a complete preliminary calculation of heat exchanger height and pressure drops. Fig. 9 provides the top view of the steam generators arrangement. There is a margin between the core radius and the steam generators to accommodate the steam lines, assumed as 8 inch pipes to assure flow speeds below 200 m/s.

Fig. 10 shows an integrated view of the reactor coolant system. The control rods drive is assumed as a rack pinion system and the rack is filled with neutron absorber and travels inside the core. This

Table 3
Values obtained for the 240 MW core.

Nuclear reactor characteristics	Value	Unit
Number of fuel elements	137	
Number of tubes per fuel element	289	
Percentage of fuel rods	90.25%	
Tubes external diameter	0.0098	m
Fuel rods length	2.11	m
Element width	0.2207	m
Core diameter	3.04	m
Minimum departure from nucleate boiling ratio at hot channel	2.05	
Output temperature	323.7	°C
Mean temperature variation in core	25.7	°C
Hot leg subcooling margin	22.1	°C
Coolant mass flow	1850	kg/s
Pressure loss at reactor	4922	Pa

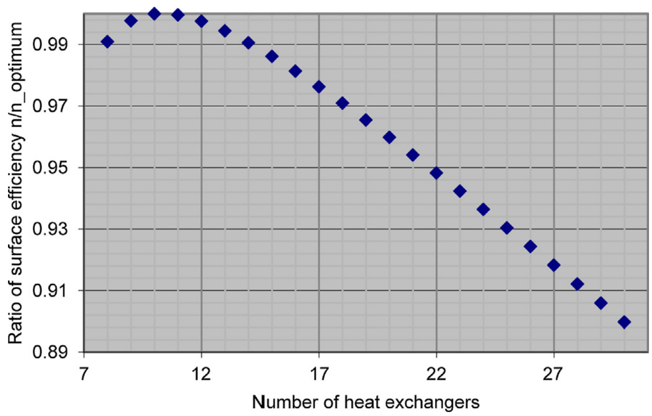

Fig. 8. Arrangement efficiency of the number of heat exchangers in function of the number of heat exchangers.

way the total height required for the pressure vessel is reduced by the travel of the control rods racks. Other compact solutions for driving the control rods may also be thought, like a gear and chain system.

Table 4 provides one solution to have a balanced steam generator with the core in terms of pressure drop. It was calculated the heat transfer coefficients using Gungor and Winterton (1986) correlation and temperatures in the secondary side, as shown in Figs. 11 and 12. It was found that the algebraic turbulence model for annular flow (Cioncolini and Thome, 2012a,b) predicted heat transfer coefficients 86% greater than those predicted by Gungor and Winterton (1986) correlation for the secondary side. However, there was little change in the global heat transfer coefficient because the main heat resistance is the thermal resistance of the Ti-3Al-2.5V plate. As a conservative approach, it was decided to adopt the smaller heat transfer coefficient of the correlation for flat tubes from Gungor and Winterton (1986) correlation.

It was investigated the critical heat flux using Zhang (2006) correlation and found that the heat fluxes found in the steam

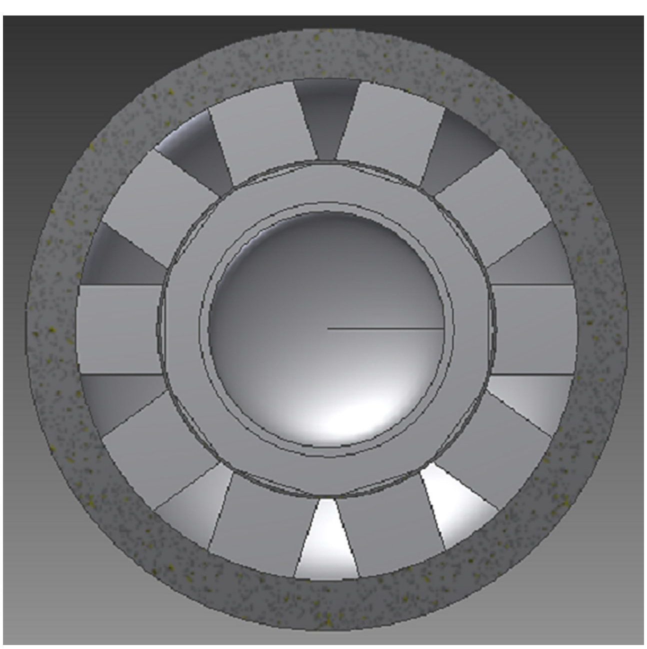

Fig. 9. Core and steam generators arrangement.

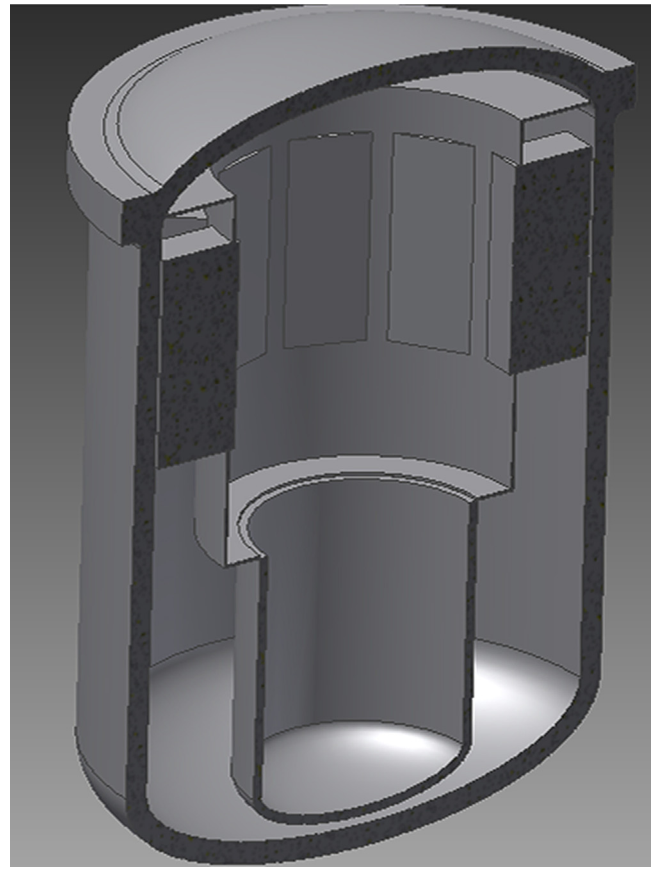

Fig. 10. Lateral view of the reactor.

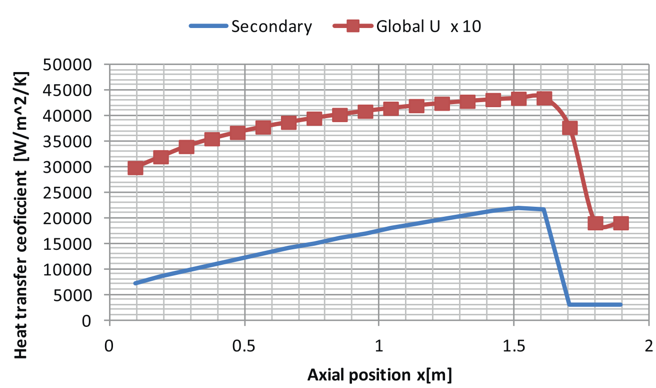

Fig. 11. Calculated global heat transfer coefficient as a function of axial position. The origin is the primary inlet and secondary outlet.

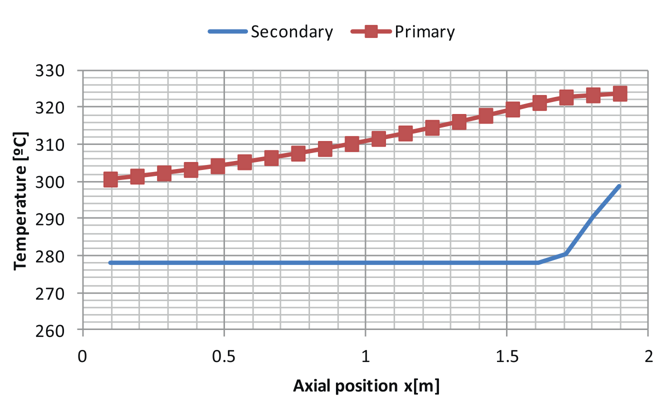

Fig. 12. Calculated primary and secondary fluids temperatures as a function of axial position. The origin is the primary inlet and secondary outlet.

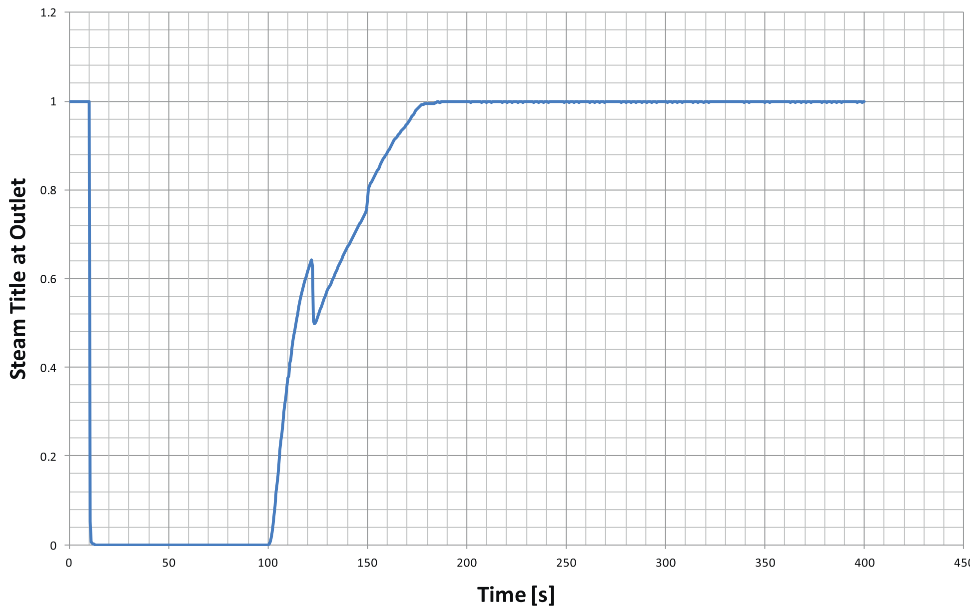

Fig. 13. Steam quality at the outlet of the steam generator.

generator are 3 orders of magnitude below the critical heat flux, which probably exclude this problem.

Table 4 summarizes the characteristics obtained for the plate heat exchangers along with some reactor coolant system parameters.

The RELAP simulation starts with the steam generator with dry steam, at the interval $T = 1$ s to $T = 10$ s, the saturated water starts to increase the flow from 0 to the nominal flow for a single channel (0.1366 kg/s). At the interval $T = 100$ s to $T = 200$ s the pressure starts to drop from saturation pressure at hot leg temperature (111 bar) down to the nominal pressure (64.5 bar). Fig. 13 presents the evolution of the steam title at steam generator outlet. It can be noted the steam title goes to 1, which means the heat transfer area is enough to assure complete vaporization of the water.

Fig. 14 shows the total heat transfer for one steam generator channel. For a total 240 MW, having 1130 channels, there must be a heat exchange of 212.4 kW per channel. According to the RELAP calculation, the average heat transfer was about 227.3 kW. That means the steam generator could be 7% smaller than first estimation.

Fig. 15 presents the charge loss, which was quite similar to the estimated charge loss using Quibén et al. (2009) correlation for flat tubes. The estimated charge loss using Quibén et al. (2009) correlation for flattened tubes was 0.677 bar and the RELAP charge loss was 0.72 bar.

Fig. 16 presents the evolution of pressure loss on primary side. It was estimated 14,787 Pa but the RELAP mean pressure loss was about 8848 Pa. This deviation may be explained by the primary fluid contraction which was not taken into account in the hand calculation.

The results of stress analysis using a finite element model are shown in Fig. 17. It can be noted that all points are below the allowable stress of 140 MPa, which demonstrate the feasibility from

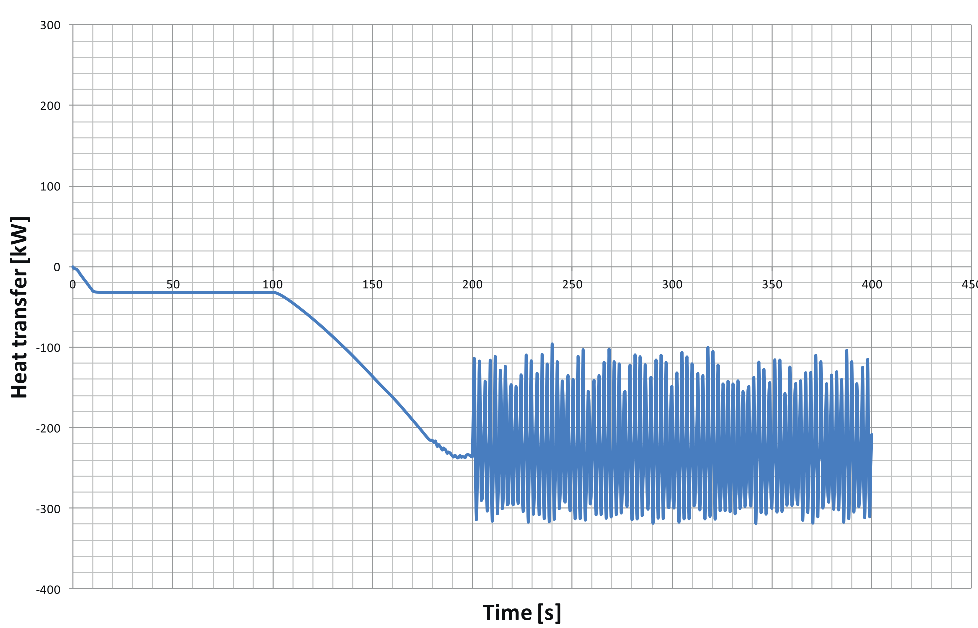

Fig. 14. Evolution of heat transfer in the steam generator with time.

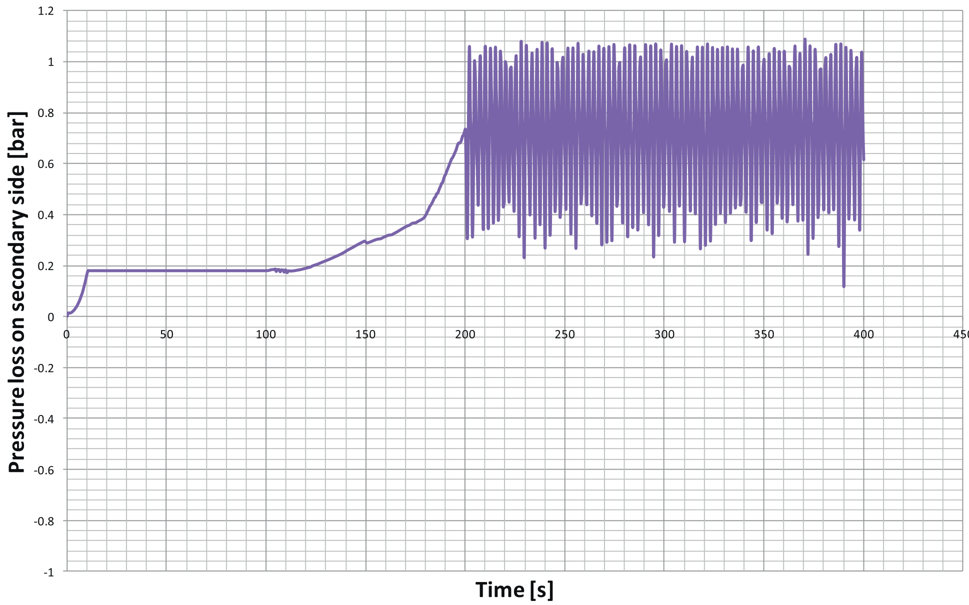

Fig. 15. Evolution of pressure loss in the steam generator with time.

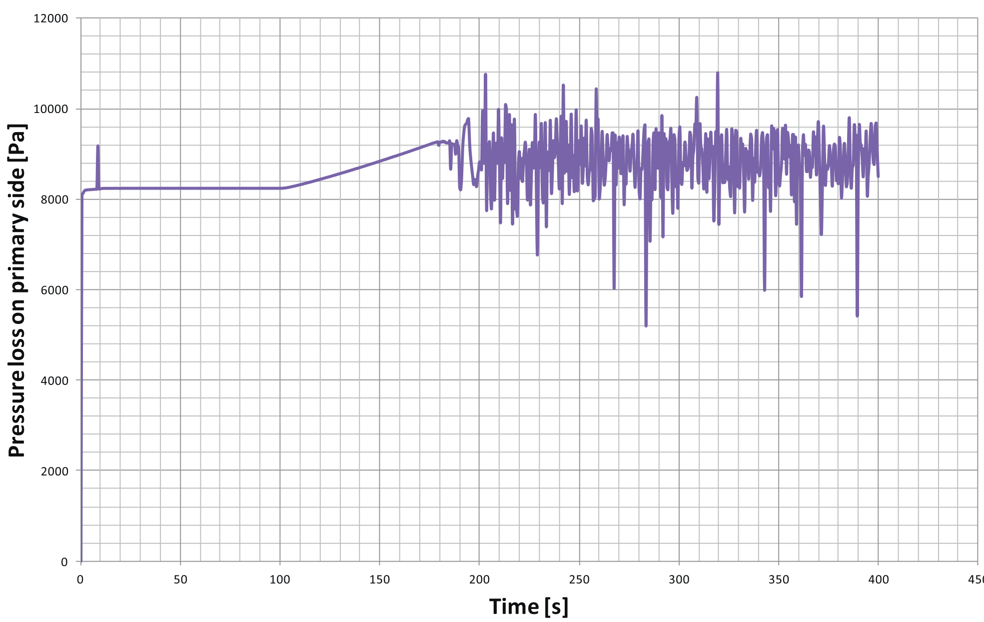

Fig. 16. Pressure loss on primary side.

mechanical point of view, despite the huge difference in pressure between secondary and primary circuits.

6. Discussion

Those results are not best estimate calculations. They are conservative calculations to allow demonstration of feasibility.

The scope of this work was only the thermal hydraulic and structural aspects. There are other aspects, like chemistry, fabrication, inspection in service. It is assumed those aspects should pose no unsolvable problem once the proposed solution is modular and applies solutions already available in the market.

From fabrication point of view, it can be stated that it should not pose problems, as there are many plate and shell heat exchangers suppliers. The suppliers state that it is possible to work with plates with thickness up to 1 mm and titanium is a typical material for the plates. The size of the plates should not be a problem as there are commercial solutions with 1.55 m^2 plates. The only additional cost would be to make a dedicated punch tool to forge the plates.

Chemistry should not also pose problems as titanium is resistant to corrosion. On the other hand, like in Babcock & Wilcox once-through steam generators, the secondary water chemistry has limitations because the solid particles cause corrosion on turbine blades.

It may not be possible to fully inspect the plates, but it is certainly possible to inspect the state of the plates using proper tools and the modular architecture eases the identification of the position of leakages. For instance, if there is a leakage, it is possible to identify the leaky heat exchanger by pressurizing the steam line with nitrogen while the vessel is open to atmosphere and filled with

Table 4
Characteristics obtained for the array of plate heat exchangers for steam generation.

Steam generator array	Value	Unit
External annulus diameter	5.14	m
Internal annulus diameter	3.64	m
Assumed secondary fluid mass flow	154.36	kg/s
Number of heat exchangers	10	
PHE width	0.857	m
PHE depth	0.714	m
PHE length	2.201	
Number of cells of plate, sec, plate, prim	1130	m
Primary heated perimeter	1550	m^2
Primary cross section area	2.324	m/s
Mean primary coolant flow speed	1.1318	
Primary coolant Reynolds number	56,443	$W/m^2/K$
Primary heat transfer coefficient	10,805	W/m^2K
Secondary heated perimeter	1132	m^2
Secondary cross section area	0.566	m/s
Subcooled secondary water flow speed	0.362	
Subcooled secondary water Reynolds	4958	$W/m^2/K$
Subcooled secondary water h	5617	$W/m^2/K$
Conduction heat transfer coefficient	11,000	W/m/K
Global heat transfer conductivity per length	3769	W/m/K
Friction factor on primary side	0.0447	
Primary pressure loss	14,787	Pa
Secondary side pressure loss	0.677	Bar

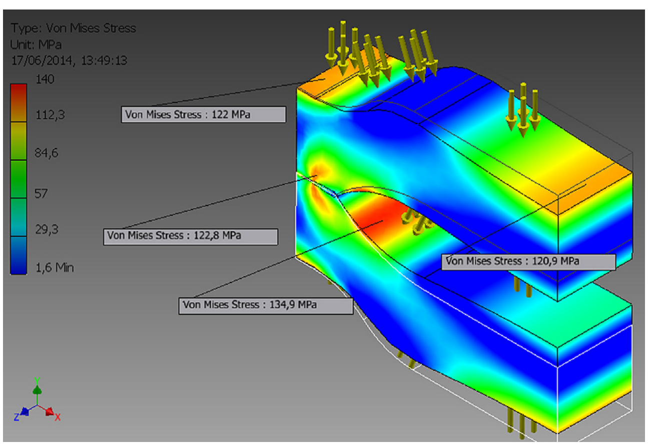

Fig. 17. Stress analysis on a half channel of the plate heat exchanger.

coolant. The heat exchanger with the leak should release bubbles which are easily noted by operators. Then it is simple to replace an entire heat exchanger.

7. Conclusions

The use of plate and shell heat exchangers for steam generation in a reactor vessel is demonstrated feasible and more compact than other solutions. Because of this compactness it is possible to have functional segregation between the power conversion function and emergency secondary heat removal function, which allows employing industrial standards in the power conversion function and make the emergency secondary heat removal function more reliable by avoiding corrosion wear during the service life.

However, as the power conversion function is a part of the first level of defense in depth, it shall be able to assure it does not cause accidents frequently. This is achieved by keeping the steam generators plates under compression, so there are not crack propagation and pipe ruptures. Only leakages and collapses may occur, which, if frequent, do not cause safety problems, but only plant availability problems.

Further work may include the calculation of pressure loss and heat transfer using more advanced tools, calculation of structural feasibility using a set of different materials and experiments to check the chemical resistance of this kind of steam generator and its reliability.

Another aspect for future work is the sizing and modeling of the safety steam generators, along its application in the design basis accidents.

References

Carlton, J., 2011. A New Nuclear Dawn? A Carbon Free Solution to Merchant Ship Propulsion. IMarEST, Southampton.
Cheng, S.K., Todreas, N.E., 1985. Hydrodynamic models and correlations for wire-wrapped LMFBR bundles and subchannel friction factors and mixing parameters. Nucl. Eng. Des. 92, 227.
Cioncolini, A., Thome, J.R., 2012a. Entrained liquid fraction prediction in adiabatic and evaporating annular two-phase flow. Nucl. Eng. Des. 243, 200–213.
Cioncolini, A., Thome, J.R., 2012b. Algebraic turbulence modeling in adiabatic and evaporating annular two-phase flow. Int. J. Heat Fluid Flow (Elsevier) 32 (2011), 805–817.
Eletronuclear, 2014. Angra 2 – Informações Gerais/Histórico/Dados Básicos, http://www.eletronuclear.gov.br/AEmpresa/CentralNuclear/Informa%C3%A7%C3%B5esAngra2.aspx (accessed 2014).
Eyring, V., 2010. Transport impacts on atmosphere and climate: shipping. Atmos. Environ. 44, 4735–4771.
Gravina, J., et al., 2012. Concepts for a Modular Nuclear Powered Containership. Elsevier, Ocean Engineering, pp. 1627–1636.
Gungor, K.E., Winterton, R.H.S., 1986. A general correlation for flow boiling in tubes and Annuli. Int. J. Heat Fluid Flow (Elsevier) 29, 351–358.
Ippolito Jr., T.D., (Master thesis) 1987. Effects of Variation of Uranium Enrichment on Nuclear Submarine Reactor Design. B.S.N.E., University of Lowell, Lowell, MA, pp. 257.
Jenkins, V., 2011. Risk and Classification Rules for Nuclear Powered Ships. Ship Power Forum. IMarEST, London.
MAN Diesel A/S, 2008. Propulsion Trends in Container Vessels. MAN Diesel A/S, Copenhagen, http://www.mandiesel.com (accessed 2014).
Quibén, J.M., et al., 2009. Flow boiling in horizontal flattened tubes: Part I - Two-phase frictional pressure drop results and model. Int. Heat Fluid Flow (Elsevier) 52, 3634–3644.
Reddy, D.G., Fighetti, C.F., (EPRI Report NP-2609) 1983. Parametric Study of CHF Data, Vol. 2 Generalized Subchannel CHF Correlation for PW&PWR Assemblies. Electric Power Research Inst., Palo Alto, CA.
Revellin, R., Mishima, K., Thome, J.R., 2009. Status of prediction methods for critical heat fluxes in mini and microchannels. Int. J. Heat Fluid Flow (Elsevier) 30, 983–992.
Royal Academy of Engineering, 2013. Future Ship Powering Options Exploring Alternative Methods of Ship Propulsion. Royal Academy of Engineering, United Kingdom.
Scientech Inc., 1999. RELAP5/MOD3 Code Manual Volume II: Appendix A – Input Requirements., pp. 194.
Tibiriçá, C.B., Ribatski, G., Thome, J.R., 2012. Saturated flow boiling heat transfer and critical heat flux in small horizontal flattened tubes. Int. J. Heat Mass Transfer 55, 7873–7883.
Tong, L.S., 1967. Prediction of departure from nucleate boiling for an axially non-uniform heat flux. J. Nucl. Energy 21, 241.
ViFlow, 2014. Laser Welded Plate Heat Exchangers Type XPS, downloaded at http://www.viflow.fi/uploads/ViFlow_GESMEX_GB_HiRes.pdf (accessed 2014).
Wade, K.C., 1995, August. Steam Generator Degradation and Its Impact on Continued Operation of Pressurized Water Reactors in the United States. Energy Information Administration/Electric Power Monthly.
Weisman, J., Ying, S.H., 1985. A theoretically based critical heat flux prediction for rod bundles at PWR conditions. Nucl. Eng. Des. (North-Holland, Amsterdam) 85, 239–250.
Yoon, S.H., No, H.C., Kang, G.B., 2014. Assessment of straight, zigzag, S-shape, and airfoil PCHEs for intermediate heat exchangers of HTGRs and SFRs. Nucl. Eng. Des. 270, 334–343.
Zhang, W., et al., 2006. Correlation of critical heat flux for flow boiling of water in mini-channels. Int. J. Heat Mass Transfer (Elsevier) 49, 1058–1072.