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ROBUSTNESS OF THE STANDARD SMART100 DESIGN TO CRITICAL HEAT FLUX DEGRADATION FOR FLOATING NUCLEAR POWER PLANT APPLICATION

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Maritime decarbonisation and restoration of Ukraine's nuclear power generation motivate small modular reactor deployment. A floating nuclear power plant concept based on two certified SMART100 units without thermal-hydraulic modifications received ABS Approval in Principle in December 2025. Thermal-hydraulic and neutronic calculations and a parametric robustness analysis under critical heat flux degradation from ship motion are presented. A robustness indicator δ_{crit} – the degradation level at which the departure from nucleate boiling ratio (DNBR) reaches the regulatory limit – is proposed. The calculated $\delta_{crit} = 63.1\%$ exceeds the literature maximum (45%) by 18%, justifying SMART100 application without modifications.

INTRODUCTION

Decarbonisation of the energy sector, restoration of Ukraine's nuclear power generation destroyed as a result of the full-scale war, and global initiatives for maritime transport decarbonisation make the search for flexible small- and medium-power nuclear solutions highly relevant. Small Modular Reactor (SMR) projects are considered by the International Atomic Energy Agency [1] and the OECD Nuclear Energy Agency [2] as a key technological direction for the coming decades.

The South Korean SMART100 (System-integrated Modular Advanced ReacTor 100) project, developed by the Korea Atomic Energy Research Institute (KAERI), is an upgraded version of the original SMART reactor (which received the world's first SMR design certification in 2012) with increased thermal power of 365 MW and fully passive safety systems in accordance with current IAEA requirements [3]. Parameters of the standard SMART100 design are addressed in [4].

In December 2025, Samsung Heavy Industries in cooperation with KAERI received Approval in Principle from the American Bureau of Shipping (ABS) for a floating nuclear power plant concept (Floating SMR, FSMR) based on two certified SMART100 units without modifications to thermal-hydraulic parameters [5]. This makes an independent quantitative assessment of the sufficiency of SMART100 thermal-hydraulic margins under ship motion conditions relevant.

A review of thermal-hydraulic characteristics of nuclear reactors under ocean conditions is presented in [6]; a detailed safety analysis of a 300-MW floating nuclear power plant in marine environment is given in [7]. The development and validation of a computational code for marine reactors is described in [8], and of the TASS/SMR-S code for SMART simulation under multi-dimensional motion in [9]. A commercial marine PWR with extended fuel cycle for container vessel propulsion is described in [10]. Experimental and theoretical studies of natural circulation and Critical Heat Flux (CHF) in rod bundles under ship motion conditions are reported in [11–13].

In the theoretical study [11], for the rod bundle geometry of a floating PWR under natural circulation in ship motion conditions, it is shown that CHF strongly depends on the period and amplitude of angular acceleration of ship motion. Literature estimates of maximum CHF degradation under ship motion conditions reach ~45% relative to stationary-condition values [11–13].

Despite the existence of detailed SMART100 calculations using the TASS/SMR-S code [4, 9] and safety analyses of floating nuclear power plants [7], a quantitative assessment of the robustness of the Minimum Departure from Nucleate Boiling Ratio (MDNBR) of the standard SMART100 design to CHF degradation modelling marine operating conditions is absent from the open literature. The objective of this work is to perform such an assessment based on thermal-hydraulic and neutronic calculations of SMART100 in the FSMR configuration with standard design parameters, and on a parametric analysis of MDNBR dependence on the CHF degradation level with determination of the critical level δ_{crit} corresponding to the regulatory limit of the W-3 (Tong) correlation [4, 15].

1. OBJECT OF STUDY AND INPUT DATA

SMART100 is an integral Pressurized Water Reactor (iPWR) with a thermal power of 365 MW and an electrical power of approximately 110 MW. The Samsung Heavy Industries / KAERI floating nuclear power plant project [5] employs two certified SMART100 units without modifications to thermal-hydraulic parameters. Therefore, the input data for the calculations in this work are the standard design parameters of SMART100 [4], summarised in Table 1.

All primary-circuit equipment (core, eight once-through steam generators with helical tubes, four main coolant pumps with canned-motor drives, and a pressuriser) is integrated within a single reactor pressure vessel, which fundamentally eliminates the possibility of a large-break loss-of-coolant accident (LB-LOCA).

Table 1

SMART100 parameters in the FSMR configuration [4, 5]

Parameter	Value
Thermal power Q, MW	365
Primary circuit pressure, MPa	15.0
Coolant inlet temperature, °C	296.0
Coolant outlet temperature, °C	322.0
Core height H, m	2.0
Number of fuel assemblies × format	57 × (17×17)
Coolant mass flow rate, kg/s	2428.7
Fuel enrichment (average), %	4.1

The core contains 57 fuel assemblies (FAs) in the standard 17×17 format with UO_2 ceramic fuel with enrichment below 5%. SMART100 safety is ensured by passive systems (PRHRS, PSIS, CMT) with an autonomous operation time of ≥ 72 h without operator action [4]. For marine applications, the PRHRS can be adapted using seawater as the ultimate heat sink [14]. The core fuel cycle is approximately 900 effective full-power days.

2. THERMAL-HYDRAULIC CALCULATION METHODOLOGY

Calculations are performed for the most thermally loaded fuel assembly at seven axial points along the core height using coolant properties from IAPWS-IF97 [16] following the methodology of [15]. The adopted radial power peaking factor K_r = 1.786 and integral safety factor K_t = 2.17 account for radial and axial power non-uniformity, manufacturing tolerances, and uncertainties in heat-transfer correlations [15].

The coolant enthalpy distribution along the core height (cosine profile accounting for effective reflector additions) [15]:

$$h_z = h_{in} + (\Delta h \cdot K_t / 2) \cdot [1+\sin(\pi c \cdot z)/\sin(\pi c \cdot H/2)], \quad (1)$$

where $\Delta h = h_{out} - h_{in}$ is the average coolant enthalpy rise in the core; c is the reflector coefficient; z is the axial coordinate measured from the core centre.

The heat transfer coefficient by the Mikheev correlation for turbulent flow:

$$Nu = 0.021 \cdot Re^{0.8} \cdot Pr^{0.43}, \quad \alpha = Nu \cdot \lambda / d_{eq}. \quad (2)$$

When the cladding temperature exceeds the saturation temperature t_s, the heat transfer coefficient is recalculated to account for subcooled nucleate boiling using the onset-of-nucleate-boiling heat flux and the reduced heat transfer coefficient α_{red} [15].

The critical heat flux (CHF) for a non-uniform axial heat flux distribution is calculated using the W-3 Tong correlation [17], which for a uniform heat flux has the form:

$$q_{cr},EU = \{(2.022 - 0.06238 \cdot P) + (0.1722 - 0.01427 \cdot P) \cdot \exp[(18.177 - 0.5987 \cdot P) \cdot X_e]\} \times \{(0.1484 - 1.596 \cdot X_e + 0.1729 \cdot X_e \cdot |X_e|) \cdot G/10^6 + 1.037\} \times \{1.157 - 0.869 \cdot X_e\} \times \{0.2664 + 0.8357 \cdot \exp(-124.1 \cdot D_e)\} \times \{0.8258 + 0.341 \cdot 10^{-6} \cdot (h_f - h_{in})\}, \quad (3)$$

where P is pressure, MPa; X_e is the equilibrium quality; G is mass flux, kg/(m²·s); D_e is equivalent diameter, m; h_f, h_{in} are saturation and inlet enthalpies, J/kg. For a non-uniform axial profile, q_{cr} is corrected by the Tong F-factor multiplier [15, 18].

The Departure from Nucleate Boiling Ratio (DNBR) is defined as:

$$DNBR(z) = q_{cr}(z) / q_z(z), \quad (4)$$

where $q_z(z)$ is the local heat flux at height z. The minimum value of DNBR along the core height is denoted as MDNBR.

The NRC safety criterion for PWR design regimes is MDNBR ≥ 1.30 [4]. This criterion is regulatory and applies to CHF calculation results obtained with any validated correlation, including W-3 [4]. In addition, a conservative value of MDNBR ≥ 1.50 (academic criterion [15]) is used in this work to check the sensitivity of the conclusions.

For the standard SMART100 pressure (P = 15 MPa), the formal upper validation limit of the W-3 correlation is 13.8 MPa; however, in international practice, W-3 is the de facto standard for all PWR types at 15…16 MPa [4]. The extrapolation uncertainty of W-3 at 15 MPa is offset by the conservative safety factor K_t = 2.17, which accounts for radial and axial power non-uniformity, manufacturing tolerances, and uncertainties in heat-transfer correlations [15].

The hydraulic resistance of the core is determined by the sum of friction losses, local resistances, hydrostatic pressure rise, and flow acceleration [15].

3. NEUTRONIC CALCULATION METHODOLOGY

Calculations are based on the two-zone Wigner-Seitz cell model and a four-group system of neutron constants [15], using the methodology adapted for marine nuclear installations [18]. The infinite-medium multiplication factor is expressed in the four-factor form:

$$k_\infty = \eta \cdot \varepsilon \cdot p \cdot \theta, \quad (5)$$

where η is the average number of secondary neutrons per absorption event; ε is the fast fission factor; p is the resonance escape probability; θ is the thermal utilisation factor. The effective multiplication factor for a finite reactor:

$$k_{eff} = k_\infty \cdot \exp(-B^2\tau) / (1 + B^2L^2), \quad (6)$$

where B^2 is the geometric buckling; τ is the neutron age; L^2 is the squared diffusion length. The fuel cycle is determined through depletion of ^{235}U and accumulation of ^{239}Pu, ^{135}Xe, ^{149}Sm with stepwise recalculation of k_∞ and k_{eff} [15].

4. RESULTS OF STANDARD OPERATING REGIME ANALYSIS FOR SMART100

Heat balance verification: at the stated flow rate G = 2428.7 kg/s and Δh = 150.29 kJ/kg (from IAPWS-IF97 for T_{in} = 296 °C, T_{out} = 322 °C at P ≈ 15 MPa), the calculated power Q = G·Δh = 365.0 MW agrees with the declared SMART100 thermal power within 0.01%. The volumetric power density q_v = 73.8 MW/m³ (26% lower than in WWER-1000); coolant inlet velocity is

2.14 m/s; coolant temperature rise across the core is 26 °C. The calculated temperature distributions along the height of the most thermally loaded fuel assembly are presented in Table 2.

Table 2

Temperatures along the core height of SMART100

No	t_{cool}, °C	t_{clad}, °C	$t_{fuel.surf}$, °C	t_{centre}, °C
1	296.0	304.7	356	450
2	301.3	335.9	543	1092
3	311.7	349.3	680	1784
4	324.0	365.1	736	2071
5	335.0	370.0	686	1793
6	341.5	363.5	570	1131
7	341.4	346.9	401	498

The maximum outer cladding surface temperature $t_{clad,max}$ = 370.0 °C (point 5) is below the allowable limit for zirconium alloys. The maximum temperature at the centre of the fuel pellet $t_{fuel,max}$ = 2071 °C (point 4) is significantly below the UO_2 melting point (≈ 2700 °C). The hydraulic pressure drop across the core is ΔP = 0.04 MPa.

The results of departure from nucleate boiling ratio calculations using the W-3 correlation are presented in Table 3.

Table 3

MDNBR along the core height for the standard regime

No	q_z, kW/m²	q_{cr}, kW/m²	MDNBR
1	152	7593	49.98
2	606	6839	11.28
3	924	5558	6.01
4	1037	4252	4.10
5	920	3242	3.52
6	601	2615	4.35
7	150	2346	15.66

The minimum MDNBR = 3.52 occurs at point 5 (above the core centre). This value substantially exceeds the NRC regulatory criterion (MDNBR ≥ 1.30) [4] and corresponds to the typical design margin of modern PWRs (MDNBR_design > 2.0).

Results of the neutronic calculations: neutron gas temperature T_{ng} = 768.7 K; infinite-medium multiplication factor k_∞ = 1.370; effective multiplication factor at the beginning of the fuel cycle k_{eff} = 1.332. The total control rod worth was not evaluated in this work – its determination requires a separate calculation. Fuel burnup calculations accounting for accumulation of fission products and transformations of ^{239}Pu, ^{135}Xe, ^{149}Sm show that with an average initial enrichment of 4.1%, the reactor sustains the design fuel cycle of 900 effective full-power days (≈ 30 months) while satisfying the IAEA enrichment constraint of < 5%.

5. MDNBR ROBUSTNESS ANALYSIS UNDER CHF DEGRADATION FROM SHIP MOTION

In the operation of floating nuclear installations (Floating Nuclear Power Plant, FNPP/Floating SMR, FSMR), inertial loads from ship motion affect the natural circulation of the coolant [8] and the critical heat flux (CHF) [11–13]. The most relevant study for the rod bundle geometry of SMART100 is [11], in which a code is developed for CHF calculation in a natural-circulation system of a floating PWR under ship motion. The work demonstrates that CHF strongly depends on the period and amplitude of angular acceleration of ship motion. Experimental studies of heat transfer in rod bundles under ship motion [13] and of coupled neutronic – thermal-hydraulic phenomena under ship motion [12] confirm that the maximum CHF degradation under ship motion can reach ~45% relative to stationary-condition values, which is adopted in this work as a conservative upper bound.

To assess the robustness of SMART100 to CHF degradation, a parametric analysis of the MDNBR dependence on the degradation level δ is performed:

$$\mathrm{MDNBR}(\delta) = \mathrm{MDNBR}_0 \cdot (1 - \delta/100), \qquad (7)$$

where MDNBR_0 = 3.52 is the baseline value (see point 5, Table 3); δ is the CHF degradation level, %. Scan results for the range δ = 0…70% are presented in Table 4.

Table 4

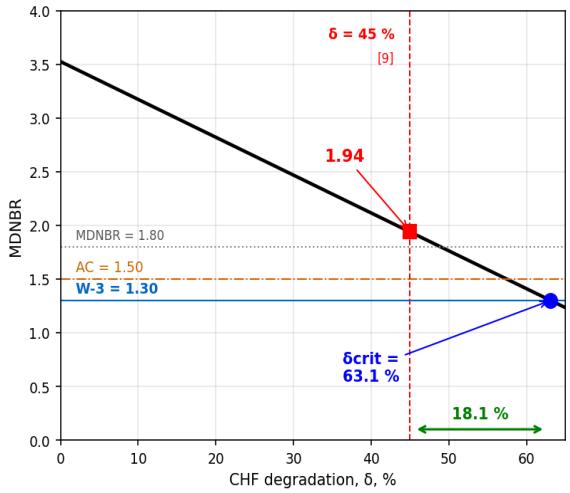
MDNBR dependence on CHF degradation

δ, %	q_{cr}, kW/m²	MDNBR	Margin over W-3, %
0	3242	3.52	+170.9
10	2918	3.17	+143.8
20	2594	2.82	+116.7
30	2269	2.47	+89.6
40	1945	2.11	+62.5
45	1783	1.94	+49.1
50	1621	1.76	+35.8
55	1459	1.59	+22.1
63.1	1199	1.30	0.0 (W-3 limit)
70	973	1.06	−18.5 (violated)

The critical CHF degradation level at which the NRC regulatory criterion (MDNBR = 1.30) [4] is reached:

$$\delta_{crit} = (1 - \mathrm{MDNBR}_{limit}/\mathrm{MDNBR}_0)\cdot 100\% = 63.1\%. \qquad (8)$$

For the alternative criterion – the conservative academic limit (MDNBR ≥ 1.50 [15]) – δ_{crit} = 57.4%. The graphical dependence MDNBR(δ) with indication of regulatory levels and the literature maximum of CHF degradation is shown in Figure.



MDNBR dependence on CHF degradation for the standard SMART100 operating regime

Key results of the analysis:

1) at the conservative CHF degradation of 45% (literature maximum from [11–13]), MDNBR = 1.94, exceeding the NRC regulatory criterion (1.30) by 49% and the conservative academic criterion of 1.50 by 29%;

2) the critical degradation level δ_{crit} = 63.1% exceeds the literature maximum of 45 by 18.1%, which corresponds to a 40% relative margin;

3) the margin is preserved even for stricter regulatory MDNBR values.

It should be noted that the 45% estimate is the upper bound of literature data. For the rod bundle geometry of SMART100 with forced circulation at a mass flux of 1700…2000 kg/(m^2·s), according to the results of [11–13], the actual CHF degradation is expected to be substantially lower. This additionally increases the real safety margin.

To justify the adopted value of 45%, a comparative analysis of literature estimates of the ship-motion effect on CHF is presented in Table 5.

Table 5

Literature estimates of ship-motion effect on CHF in rod bundles

Geometry / conditions	Study type	CHF degradation, %	Source
Rod bundle of floating PWR, natural circulation	theoretical analysis	up to 45	[11]
Rod bundle, natural circulation + ship motion	experiment + analysis	regime-dependent	[12]
Rod bundle, heat transfer under ship motion	experiment	up to 30	[13]

As Table 5 shows, the available literature data for the rod bundle geometry of a PWR under ship motion confirm a maximum CHF degradation of approximately 45% from theoretical analysis [11] and up to 30% from experimental studies [12, 13]. The upper value of 45% is adopted in this work as a conservative limit, ensuring a safety margin for geometric uncertainty. If, for the actual SMART100 configuration, CHF degradation amounts to only 25…30%, the MDNBR margin over the W-3 limit will be even greater: at δ = 30%, MDNBR = 2.47, with a 90% margin over the 1.30 limit.

6. DISCUSSION

The obtained critical degradation level δ_{crit} = 63.1% provides a quantitative justification for the Samsung Heavy Industries / KAERI decision to use two certified SMART100 units without modifications to thermal-hydraulic parameters in the floating nuclear power plant project that received ABS Approval in Principle in December 2025 [5]. To reach the W-3 regulatory limit (MDNBR = 1.30), a CHF degradation of 63% is required – substantially in excess of the literature maximum of 45%.

Limitations of the work should be noted:

1) calculations were performed using a deterministic 1D methodology for the most thermally loaded fuel assembly [15]; subchannel analysis (COBRA-IV, VIPRE, COBRA-EN) is required for regulatory justification;

2) the W-3 correlation is validated for P ≤ 13.8 MPa; at P = 15 MPa, extrapolation is used, partially compensated by the increased safety factor K_t = 2.17;

3) the 45% CHF degradation is taken from [11] as a conservative upper bound; FSMR regulatory justification requires dedicated benchmarks for the PWR rod bundle geometry on ship-motion test rigs;

4) the analysis is based on SMART100 parameters from open sources [4, 15]; full verification requires access to KAERI design documentation.

Despite these limitations, the obtained margin of 18.1% over the literature maximum is substantial and confirms the conservatism of the SHI/KAERI decision. The proposed parametric MDNBR(δ) analysis methodology can be applied to other iPWR projects (NuScale US460, ACP100) for preliminary assessment of suitability for floating applications.

CONCLUSIONS

1. Thermal-hydraulic and neutronic calculations of SMART100 in the FSMR configuration have been performed with the standard design parameters [4]. The results: $t_{clad,max}$ = 370 °C, $t_{fuel,max}$ = 2071 °C, $MDNBR_{min}$ = 3.52, ΔP = 0.04 MPa, k_{eff} = 1.332, fuel cycle of 900 EFPD at an average enrichment of 4.1%. The heat balance agrees with the declared thermal power within 0.01%.

2. A parametric analysis of MDNBR robustness to CHF degradation under ship motion has been performed. At a conservative degradation of 45% (literature maximum from [11–13]), MDNBR = 1.94, exceeding the NRC regulatory criterion (1.30) by 49% and the conservative academic criterion (1.50) by 29%.

3. The critical CHF degradation level δ_{crit} = 63.1% (for the NRC criterion, MDNBR ≥ 1.30) has been determined, exceeding the literature maximum of 45% [11] by 18.1%. For the conservative academic criterion MDNBR ≥ 1.50, δ_{crit} = 57.4%.

4. The viability of the standard SMART100 design without modifications to thermal-hydraulic parameters for FSMR application has been quantitatively justified, in agreement with the Samsung Heavy Industries/KAERI decision to use two certified SMART100 units in the floating nuclear power plant project [5].

5. The proposed parametric MDNBR(δ) analysis methodology can be applied for preliminary assessment of the suitability of other iPWR projects for floating applications. Full FSMR regulatory justification requires subchannel analysis and dedicated CHF benchmarks for the rod bundle geometry under ship motion conditions.

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СТІЙКІСТЬ ШТАТНОГО ДИЗАЙНУ SMART100 ДО ДЕГРАДАЦІЇ КРИТИЧНОГО ТЕПЛОВОГО ПОТОКУ ДЛЯ ПЛАВУЧОЇ АЕС

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Декарбонізація морського транспорту та відновлення ядерної енергогенерації в Україні актуалізують застосування малих модульних реакторів. Концепт плавучої атомної електростанції на основі двох сертифікованих SMART100 без модифікацій теплотехнічних параметрів отримав ABS Approval in Principle у грудні 2025 р. Виконано теплогідравлічний та нейтронно-фізичний розрахунки штатного дизайну та параметричний аналіз стійкості до деградації критичного теплового потоку в умовах хитавиці. Запропоновано показник $\delta_{крит}$ – рівень деградації, при якому коефіцієнт запасу до кризи теплообміну досягає нормативної межі. Отримано $\delta_{крит}$ = 63,1%, що перевищує максимальну літературну оцінку (45%) на 18%, обґрунтовуючи застосування штатного SMART100 у плавучій АЕС без модифікацій.