Characteristics of Core Thermal-Hydraulic Design of SMART-P

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1. Introduction

The SMART (System-Integrated Modular Advanced ReacTor) is an integral-type advanced light water reactor which is purposed to be utilized as an energy source for sea water desalination as well as a small scale power generation. A prototype of this reactor, named SMART-P, has been studied at KAERI in order to demonstrate the relevant technologies incorporated in the SMART design. Due to the closed-channel type fuel assemblies and low mass velocity in the reactor core, the thermal hydraulic design features of SMART-P revealed fairly different characteristics in comparison with existing PWRs. The allowable operating region of the core, from the aspect of the thermal integrity of the fuel, should be primarily limited by two design parameters; critical heat flux (CHF) and fuel temperature. The occurrence of CHF may cause a sudden increase of the cladding temperature which eventually results in the fuel failure. The fuel temperature limit is relevant to a fuel failure mechanism such as a fuel centerline melting or a phase change of metallic fuels. Two phase flow instability is also an important design parameter since a flow oscillation may trigger a CHF or mechanical vibration of the channel. The characteristics of important thermal-hydraulic design parameters have been investigated for the SMART-P core with the closed-channel type fuel assemblies which contained non-square arrayed SSF (Self-sustained Square Finned) fuel rods.

2. Core TH Design Characteristics

2.1 Critical heat flux

A series of CHF experiments has been conducted for the SSF rod bundles in water and Freon test facilities. The effects of the heated length, the axial and radial power distributions, the unheated rods, and the bundle size were investigated as described in Table 1. The parametric ranges of the experiments covered a pressure from 6 to 18 MPa and a mass flux from 200 to 3000 kg/m^2/s. A CHF correlation was developed for the tightly spaced rod bundle on the basis of the bundle-averaged conditions at the axial location of the CHF occurrence. Referring to the EPRI correlation, the basic form of the correlation equation was selected as,

$$q''_{CHF} = \frac{A \times C_1 - \chi_{in}}{B \times C_2 + \dfrac{(\chi_c - \chi_{in})}{q''_{loc}} \times C_3} \qquad (1)$$

where, $A = a_1 P_r^{a_2} G_r^{a_5 + a_7 P_r}$, $B = a_3 P_r^{a_4} G_r^{a_6 + a_8 P_r}$

The correction factors, C_1, C_2, and C_3, accounted for the effects of the non-uniform axial power shapes and the channel geometry on the CHF. The coefficients of the correlation were empirically determined from the least square fitting of the CHF data for the tightly spaced test bundles. As the result of M/P analysis, the tolerance limit of the CHF correlation was determined as 1.16.

Table 1. Description of CHF test bundles

Parameters	Values
Number of TS	14
Number of rods / TS	7, 19, 55
Number of unheated rods/TS	0, 1, 2
Number of data, water/Freon	601 / 1092
Heated length, m	0.4, 0.8
Axial power shapes	Uniform, Cosine, Top/Bottom peak
Radial power shapes	Uniform, Nonuniform

2.2 Subchannel code and thermal mixing

A subchannel analysis code, MATRA-SR[1], was employed to calculate the enthalpy and flow distributions inside the test bundles. The turbulent mixing model was introduced in the subchannel code in order to account for the exchange of energy and/or momentum between the subchannels. According to the equal-mass-exchange model, the turbulent mixing flow rate per unit length from the subchannel *i* to *j* can be expressed as

$$w'_{ij} = \beta s_{ij} G_{ij} \qquad (2)$$

where s_{ij} and G_{ij} are the gap size and average axial mass flux for the subchannels *i* and *j*. The turbulent mixing parameter (β) was determined from the analysis of thermal mixing test data for a SSF 19-rod test bundle. The optimum value of β was determined at the condition when the RMS error of the channel exit temperature differences between the measured and predicted values had its minimum value. From the analysis results for fifty different test conditions which covered pressure from 5 to 18 MPa, and mass flux from 100 to 2000 kg/m^2/s, it was found that the optimum β value can be evaluated as 0.05 for the SSF test bundle.

2.3 Pressure drop in rod bundles

It is necessary to predict the hydraulic losses in rod bundles accurately for reactor design calculations. The hydraulic characteristics of the SSF rod bundle were investigated by conducting a pressure drop test under atmospheric pressure and non-heating conditions with a full scale 55-rod test bundle. The experimental data was compared with the friction factor model for smooth round tubes and rod bundles with wire wrapped spacers. It was found that the parametric behavior of the friction loss coefficient with respect to Reynolds number was similar to the correlations for wire wrapped rod bundle. From the experimental data, a friction factor correlation for the SSF test bundle at higher Reynolds number was obtained as follows:

$$f = 0.49 \cdot R_e^{-0.275} \quad \text{,for} \quad 3500 < R_e \leq 40000 \qquad (3)$$

2.4 Two-phase flow instabilities

The onset of a density wave oscillation is highly interesting in a core which consists of multiple closed-channel type fuel assemblies. A linear stability analysis model based on the frequency response method, named ALFS (Analysis of Local Flow Stability), was developed[2] to investigate the characteristics of a flow instability in the reactor core. Experiments for the measurement of the local flow instability in two parallel channels have been conducted in a high-pressure water test facility in order to investigate the parametric effects and to obtain experimental data for the validation of the linear stability analysis model. The parametric effects of the axial power shape and unheated riser were examined by employing three different round tube test sections. The bundle effect was investigated with 19-rod SSF test section. The test range covered pressure from 6 to 16 MPa and channel power from 6 to 25 kW. The predictability and conservatism of the linear model was confirmed by comparing with the experimentally observed marginal stability boundaries (MSB). The stability margin of the reactor to the density wave oscillation was evaluated by a linear stability analysis model by accounting for the hot channel factors at the lower operating boundary of pressure.

2.5 Hot channel analysis

The SSF fuel bundles were encompassed by closed channels in order to confine the fuel assembly. For the purpose of optimizing the flow distribution in the reactor core, the fuel assembly channels in the whole core were divided into four hydraulic groups according to the channel power distribution during the lifetime of the core. The optimized flow distribution was obtained by providing different inlet orifices for each hydraulic region. The COMA (Core One-dimensional Multi-channel Analyzer) code[3] has been developed for the hydraulic profiling of the core as well as the hot channel analysis. The uncertainties of global design parameters such as core thermal power, flow rate, pressure, core inlet temperature, and correlations were accounted for the determination of the core pressure drop boundary condition. The limiting fuel assembly channels for the CHF and flow instability were investigated by applying the boundary condition of a constant pressure drop in the core. The minimum CHFR and maximum fuel temperature were calculated by considering the uncertainties of the local design parameters which included fuel fabrication tolerances and the local deviation of thermal hydraulic conditions.

2.6 Other design considerations

The variation of fuel channel geometry should be considered which could be caused by fuel swelling due to the accumulation of the fission products throughout the core life. The influence of fuel swelling may appear in (i) the increase of core pressure drop which results in the decrease of the system flow rate, (ii) the variation of core flow distribution due to the history of burnup for each fuel assembly, and (iii) the local maximum gap closure which affect the CHFR margin.

Due to the adoption of N_2 gas pressurizer, the effects of non-condensable gas should be considered on the heat transfer and CHF. It was known that the CHF reduction maybe due to the increased rate of vaporization produced by generation of additional nucleate sites on the heating wall. In the case of degassed water the number of nucleation sites increased only because of increased surface heat flux density. However, in the case of gas-containing water, the additional nucleation sites apparently cause critical conditions to occur at lower heat flux densities. The magnitude of influence depends on the gas content and flow regime as well as the channel geometry.

3. Conclusion

The thermal hydraulic design features of SMART-P core were described which was characterized by the closed-channel type multiple parallel fuel assemblies and lower RCS flow rate. Important design parameters such as CHF, subchannel analysis and thermal mixing, pressure drop and flow instability, hot channel analysis and some design considerations were discussed.

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