論文の内容の要旨

論文題目　Three-Dimensional Core Design of Large Scale Supercritical Light Water-Cooled Fast Reactor
(大型超臨界圧軽水冷却高速炉の 3 次元炉心設計)

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Supercritical water cooled reactor (SCWR) system has been regarded as an innovative reactor system for economical electricity generation due to its simplified direct steam cycle system and high thermal efficiency.

SCWR can be composed as either thermal or fast system owing to its lower coolant density in upper part of the core and its high coolability. Supercritical water-cooled fast reactor (SWFR) system can be accomplished with tighter fuel pin arrangement. Additional moderator is not necessary in the fast system. Accordingly, the supercritical fast reactor system has high power density and consequent more compact pressure vessel size. The high power density and compact size of pressure vessel gives further economical potential in aspect of capital cost reduction as well as plant simplification. Furthermore, steam cycle system and safety system is compatible with those of thermal system. Difference between thermal and fast system is the reactor core.

Fuel rod failure modes and associated fuel rod design criteria that are expected to be limiting in SWFR operating condition were established in fuel rod design study. Maximum linear heat rate of 39kW/m and fuel centerline temperature of 1900°C are used as thermal design criteria for ensuring fuel integrity. Maximum cladding surface temperature of 650°C is used to avoid cladding overheating. Flow dynamic pressure of 0.02MPa is used to avoid excessive flow induced vibration. As thermo mechanical criteria, cladding collapse and rupture are considered as design criteria.

The fuel rod design parameters are determined with those limiting fuel rod design criteria including thermo-hydrodynamic consideration and thermo-mechanical consideration and with the design goal of high outlet temperature over 500°C. The fuel rod diameter size and its P/D ratio are determined to be 7.6mm and 1.14, respectively, which is mainly limited by the consideration of excessive flow induced vibration.

Thermo-mechanical behaviors of fuel rod and cladding under irradiation have been analyzed by FEMAXI-6 fuel rod performance code based on finite element method (FEM). High coolant system pressure of 25MPa plays a positive role against PCMI and gas pressure loading, which allows smaller gap and is beneficial for reducing fuel centerline temperature.

Available fuel rod design ranges are determined in terms of gas plenum length and its initial pressure for both upper and lower gas plenum location with quantitative evaluation of creep rupture life fraction of fuel cladding. The minimum gas plenum lengths of fuel rod are predicted to be 110cm and 70cm for upper and lower gas plenum locations, respectively. In a viewpoint of mechanical strength of fuel cladding against creep rupture and cladding collapse, currently available stainless steels or being developed have a potential for application to SWFR. The mechanical strength required for SWFR fuel rod application is also quantitatively determined by fuel rod analysis.

Past core design studies had been based on 2-dimensional R-Z approximation, which yields large uncertainty in principle core performance such as average coolant outlet temperature by mismatch between assembly power and flow rate. The average coolant temperature widely varies with regard to the number of flow rate regions assumed in R-Z approximation. Local power distribution within an assembly could not be taken from those analyses. The R-Z approximation also restricts core arrangement to only radial heterogeneous core.

Three-dimensional nuclear core design procedure fully coupled with thermal hydraulic calculation, which are based on tri-z fine mesh neutron diffusion solution and single channel analysis, is developed in this study. Evaluation of pin-by-pin power distribution and equilibrium cycle search is implemented in core calculation procedure. Three-dimensional design procedure permits more flexible core arrangement for negative void reactivity and eliminates the concern of mismatching between assembly power level and flow rate, which allow more accurate evaluation of core outlet temperature and cladding surface temperature.

Large commercial scale SWFR core design is investigated for negative coolant void reactivity and high core average temperature with ZrH layer. The core arrangement for negative void reactivity is proposed. The final core has composite type arrangement with internal blanket rings and scattered loading of blanket assemblies in outer core regions. Several design method for improving core outlet temperature are also proposed in this study.

SWFR core could be composed as compact system having core average power density of 156W/cm^3 including all blanket regions, which is 1.5 times higher than that of current PWR and is 2.6 times higher than that of Super LWR. The designed core has equivalent diameter of 2.7m with similar active core height. High core average outlet temperature of 503°C is also achieved by employing radial fuel enrichment zoning in seed assembly and downward flow cooling in some portion of seed fuel assemblies while keeping design criteria of MLHGR of 39kW/m and MCST of 650°C. Assembly average discharge burnup is evaluated to be 68.3MWd/kgHM. Small pressure vessel size and simplified direct steam cycle with higher thermal efficiency and discharge burnup give an economical potential in aspect of capital and operating cost.

Subchannel analysis has been carried out for SWFR fuel assembly. Peak cladding surface temperature difference arising from coolant channel heterogeneity is calculated by using the subchannel analysis code and is evaluated to be about 18.5°C. Large subchannel heterogeneity in hexagonal fast reactor fuel assembly is controlled by altering coolant channel flow area rather than adjusting P/D ratio.

Maximum cladding surface temperature at nominal condition is evaluated to be 645.3°C considering inter-channel flow mixing and coolant channel heterogeneity over the cycle. Several design considerations can be made based on the results of subchannel analysis.

Local power peaking in upward fuel assembly should be kept below 1.2 to avoid excessive cladding surface temperature deflection and to keep conservatism of core design calculation coupled with single channel analysis. Employing downward flow in the region having high local power peaking factor is effective to suppress excessively high PCST difference between single and subchannel analyses in peripheral core regions in which seed assemblies have steep power gradient.

Statistical thermal design uncertainty associated with PCST calculation is evaluated by Monte-Carlo sampling technique combined with subchannel analysis code. Uncertainties involved in system parameter variation, nuclear enthalpy rise hot factor and engineering temperature rise hot factor are taken into account in statistical thermal design uncertainty evaluation.

Maximum thermal design uncertainty associated with MCST is evaluated to be 31°C by Monte-Carlo sampling procedure and is in a good agreement with that from RTDP method. Effect of downward flow in seed region on sensitivity is investigated by improved Monte-Carlo thermal design procedure. Employing downward flow in seed assemblies are expected to be beneficial for reducing contributions of local parameter uncertainties to total uncertainty. The downward flow cooling of seed assemblies reduce total coolant enthalpy rise in hot assembly, which reduce the sensitiveness of local parameter uncertainties.

MCST including statistical uncertainty is predicted to be 681°C (650°C for nominal + 31°C) for SWFR which ensure 95/95 limit. This value provides one of limiting thermal condition that might be occurred during normal operation within engineering uncertainty and can be used as initial condition of safety analysis and for evaluation of creep behavior or stress corrosion cracking of fuel cladding under irradiation.

The conceptual design of large scale supercritical pressure light water cooled fast reactor is carried out comprehensively with considerations of fuel rod mechanical behavior, nuclear core physics and thermal hydraulics including statistical uncertainty. High temperature operation of SWFR is feasible within thermal and mechanical design criteria which do not significantly exceed those of LMFBR design.