Physics Design Codes for NHR-200 Heating Reactor

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ABSTRACT

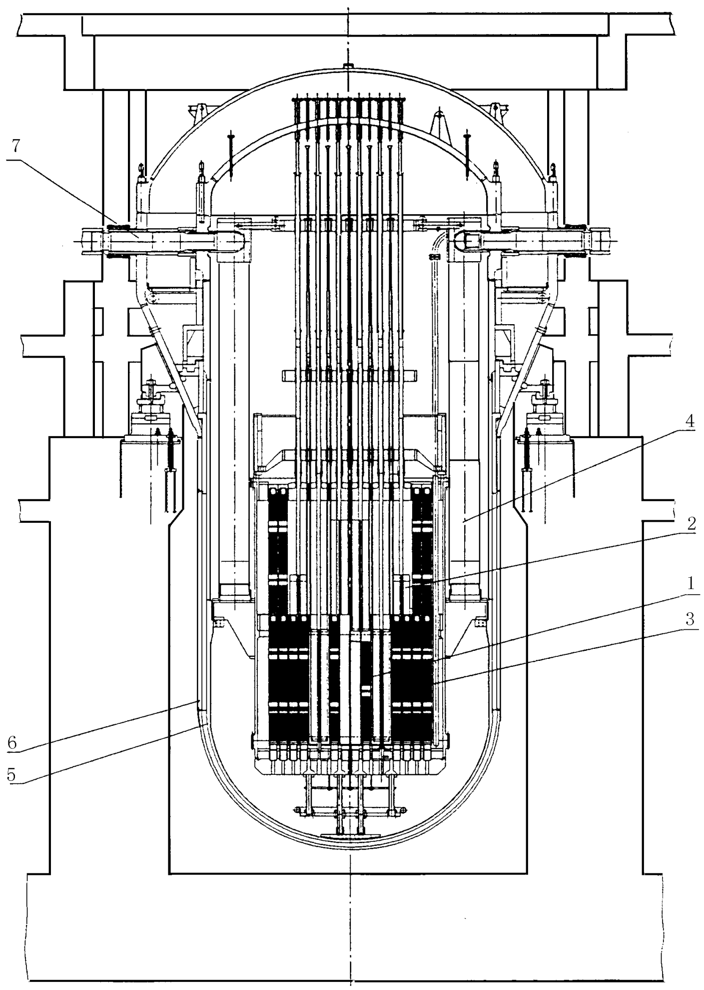
This paper describes reactor physics design codes for NHR-200 nuclear heating reactor developed by the Institute of Nuclear Energy Technology of Tsinghua University. The qualifications of these codes as design tools are assessed. The main design features of NHR-200 and its physics aspects of the reactor core have been discussed. The code development and validation work being carried out at INET of Tsinghua University for NHR-200 physics design is outlined.

1. INTRODUCTION

NHR-200 is a PWR type of low-pressure integral nuclear heating reactor with 200MWt. This particular type of reactor was designed by the Institute of Nuclear Energy Technology (INET) of Tsinghua University as an environment-friendly alternative to traditional fossil-fuel district heating systems in northern Chinese cities for meeting the growing demands in winter. Originally NHR-200 was designed for the Daqing project with 2 units of such heating reactors in early 90^{th} of the last century. This project was the first attempt to build industrial-scale demonstration nuclear heating plant in China. Unfortunately, it was cancelled near the turning of the last century because of some unexpected reasons, mainly the local government regrouping in Daqing City. However, other sites, such as Shandong province in Shandong peninsula and Shenyang in northeast of China, recently have shown great interests to shift the project to these places either for seawater desalination or district heating. Shandong province plans to apply NHR-200 for producing potable water by means of seawater desalination because of the shortage of fresh water resources along the coastal region of the peninsula. However, the authority of Shenyang intends to use nuclear district heating in winter season to attenuate the environmental burden due to burning coal for heating in urban district. The state economic planning committee has approved the preliminary assessment to the feasibility of installing 2 units of NHR-200. The detailed evaluations of the relevant projects are positively going on now in both Shandong and Shenyang. The fate of the innovative projects related to NHR-200 applications may probably be decided soon. The potential market in the coastal area of northern China, where the prices of fresh water and natural gas are rather high (the production costs could be higher than 1$/m^3 for tap water and 0.2$/m^3 for natural gas), seems promising for commercialization of NHR-200.

The main design characteristics of NHR-200 can be reflected by its simple structure and super-safety shown as Fig.1. The entire primary system is integrated within the reactor vessel. The primary coolant is driven by natural circulation due to the gravitational head of density difference between the hot leg and the cold leg. No pumps exist in the primary loop so that the probability of the accidents induced by pump failure is eliminated.

Because of its highly passive safety characteristics, the license to install NHR-200 near urban districts can be granted in China if the assessment by nuclear safety authority credits that all safety criteria are fulfilled. The licensing procedure aiming at this type of nuclear facility has been clearly specified in China. The detailed design features of NHR-200 were described before by Wang, et al. [1], here omitted.



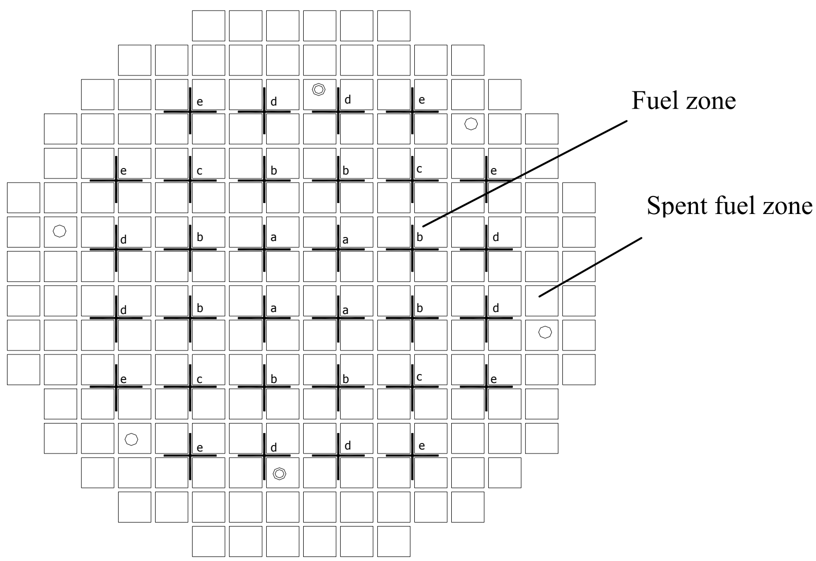
1. Core; 2. Control rod; 3. Used fuel storage shelf; 4. Main heat exchanger
5. reactor vessel; 6. Guard vessel; 7. Pipe in intermediate loop

Fig.1 Natural Circulation Nuclear Heating Reactor -NHR-200

Taking into account the renewing interests to NHR-200 in Chinese market, the importance of the computer codes for reactor physics design must be addressed. This paper describes the main physics design characters of NHR-200 and the computer code packages employed.

2. REACTOR CORE DESIGN of NHR-200

The reactor physics design for NHR-200 is the same as that for other types of light water reactors. It includes the following main aspects: selecting reactor fuel assemblies with different enrichments; defining fuel load pattern and control rod arrangement of the first core; specifying fuel cycle length; designing the worth of control rods and shut-down margin; calculating power distribution peaking factors, reactivity coefficients and neutron kinetic parameters; as well as fuel management, etc.



a group: 4; b group: 8; c group: 4; d group: 8; e group: 8

Fig.2 reactor core and control rods arrangement

The load pattern of NHR-200 core is shown in figure 2. There are 96 fuel assemblies categorized into 3 types with 1.8wt%, 2.4wt% and 3.0wt% U^{235} enrichments respectively. Each assembly contains 12x12-3 rods arranged in square pitches. Low enriched fuel assemblies are in the center zone and high-enriched ones are in the outer rim. Burnable poisons are arranged within fuel assemblies to compensate the redundant reactivity, but no boron solution is used in reactor vessel. The in-out reloading method is adopted for refueling management and each time 1/4 core is removed from the inner fuel zone to the outer spent fuel zone for in-vessel storage. With this refueling strategy, the average burn-up of fuel can increase up to 20%. The reactor physics design parameters are referenced to the specification of NHR-200 preliminary design [2]. The main design parameters of NHR-200 are listed in table 1. The design of NHR-200 is of enough safety margins for both steady state and transient operations.

Table 1 main design parameters of NHR-200

Nominal heating power	MWth	200
Reactor thermal power	MWth	200
Design lifetime of the heating plant	year	40
Pressure of the primary coolant in reactor vessel	MPa	25
Core inlet/outlet temperatures	^oC	145/210
Pressure of the intermediate loop	MPa	3.0
Inlet/outlet temperatures of the intermediate loop	^oC	95/145
Inlet/outlet temperatures of heating network	^oC	130/65
Configuration of the fuel assembly		12x12-3 (141 fuel rods)
Number of fuel assemblies of the first core		96
Number of control blade drivers		32
Material of the absorber		B_4C
Density of B_4C	g/cm^3	1.651
Enrichments in different zones of the first core	%	1.8/2.4/3.0
Length of the active zone of the reactor	m	1.9
Equivalent diameter of the active reactor core	m	1.924
Material of the burnable poison		Gd_2O_3
Concentration of the burnable poison	wt%	1%/7%
Density of Gd_2O_3	g/cm^3	10.15/9.87
Number of the main heat exchangers		4
Total heat transfer area of the main heat exchangers	m^2	2100
Loaded mass of UO_2 in reactor core	t	14.41
Loaded mass of metal U	t	12.7
Density of UO_2	g/cm^3	10.2
Core-wise average power density	kW/L	25.1
Fuel rod average linear power generation rate	W/cm	78.9
Maximum fuel rod linear power generation rate	W/cm	264.2
K_{eff} (cold, zero power, no Xe, BOC)		1.059
Power peaking factor		3.34
Average burn-up (equilibrium state)	MWD/ TU	30000
MDNBR steady state/transient		3.34/ >1.35
Fuel center maximum temperature/Safety criterion	^oC	683.3/2590

3. REACTOR PHYSICS DESIGN CODES

Codes available in INET for designing reactor physics of NHR-200 include CASMO-3/SIMULATE-3, NNGFM, ORFM and TQE/NGFM, etc. Except CASMO-3/SIMULATE-3, other codes were mostly developed by the physicists, nuclear engineers and faculty members at INET of Tsinghua University. A number of worldwide famous reactor physics design tools, such as WIM4D, CITATION and MCPN, have been used to test the validity of the codes developed by INET for NHR-200 design. The following context in this section will introduce the main functions of the codes mentioned above.

CASMO-3/SIMULATE-3 code package is developed by STUDSVIK of America, Inc. CASMO-3[3] plays the role to calculate few-group cross sections and reaction rates for any region of the fuel assembly, which are prepared for use in overall reactor calculation. CASMO-3 is based on multi-group two-dimensional transport theory and is used for burn-up calculations on BWR and PWR assemblies or simple pin cells including the capabilities to handle four BWR bundle and four PWR quarter bundles (PWR color set) cases and to generate baffle/reflector data. Two libraries containing microscopic cross section in 70 and 40 energy groups respectively attached to CASMO code package collects nuclear data ranging from 0 to 10MeV in energy spectra. SIMULATE-3[4] is an advanced two-group nodal code for the analysis of both PWRs and BWRs, in which fourth-order polynomial representations of the intranodal flux distributions in both the fast and thermal groups. The code is used for criticality search, start-up prediction, fuel management, core follow, reload physics calculations, rod worth, shut down margin, xenon transients and reconstruction of pin power. The combined code package is specially developed for PWR and BWR reactor core physics design and has been well validated among the users around the world. NHR-200 is a kind of PWR using quite standard square pitch fuel assemblies, so that CASMO-3/SIMULATE-3 is applicable to NHR-200 core physics design. Although the size of NHR-200 is relatively small with the approximate buckling value of 0.000667 cm^{-2} which is roughly as 2 to 3 times large as that of a standard 1000MWe PWR, its neutron leakage rate is only about 0.03 being slightly larger than that (~0.015) of a standard PWR. The validation of applying CASMO-3/SIMULATE-3 for NHR-200 is tolerable in view of engineering acceptance.

NNGFM[5] is a computer code for reactor core calculation based on nodal Green's function method subject to Neumann boundary condition. It was developed by INET in early 90's of the last century for reactor physics design of NHR-200. The NNGFM code applies a similar algorithm of NGFM[6] developed by Lawrence R.D. and Dorning J.J. but with different boundary condition. The 3-D neutron diffusion equation is converted into three coupled 1-D equations through transverse averaging as follows:

$$-\nabla \cdot D_g^k \nabla \Phi_g^k + \Sigma_g^k \Phi_g^k = \sum_{g'=1}^{G} [\tfrac{Xg}{Keff} \nu \Sigma_{fg'}^k + \Sigma_{g'g}^k] \Phi_{g'}^k , \quad g = 1,2,\cdots G \qquad (1)$$

Partially integrating Eq. (1) yields:

$$-D_g^k \frac{\partial^2}{\partial u^2} \Phi_{gu}^k (u) + \Sigma_g^k \Phi_{gu}^k (u) = Q_{gu}^k (u) - L_{gu}^k (u) \qquad (2)$$

Where $u = x, y, z$, $\mathrm{g} = 1, \cdots, \mathrm{G}$, $\mathrm{k} = 1, \cdots, \mathrm{K}$

The Green's function for slab geometry with the Neumann boundary condition satisfies the following equations:

$$-D_g^k \frac{\partial^2}{\partial u^2} G_{gu}^k (u,u_0) + \Sigma_g^k G_{gu}^k (u,u_0) = \delta(u - u_0) \qquad (3)$$

$$\frac{\partial G_{gu}^k}{\partial u}(u,u_0)]\bigg|_{u=\pm a_u^k} = 0$$

The flux integral equation is:

$$\Phi_{gu}^{k}(u)=[G_{gu}^{k}(u,u_0)J_{gu}^{k}(u_0)]_{u_0=-a_u^k}-[G_{gu}^{k}(u,u_0)J_{gu}^{k}(u_0)]_{u_0=a_u^k}$$
$$+\int_{-a_u^k}^{a_u^k}du_0\cdot[Q_{gu}^{k}(u_0)-L_{gu}^{k}(u_0)]\cdot G_{gu}^{k}(u,u_0) \qquad (4)$$

where $J_{gu}^{k}(u_0)=-D_g^k\frac{d\Phi_{gu}^{k}(u_0)}{du_0}$

The integral equations are approximated using a weighted residual procedure applied within each node. The resulting matrix equations, when solved in conjunction with the linear form of the nodal balance equation, provide the necessary additional relationships between the interface net currents and the flux within the node. The partial neutron flux and the neutron source term are expanded by Legendre orthodox polynomials of first three orders in each node when Eq. (4) is solved numerically.

According to continuity conditions of the heterogeneous fluxes and net currents on the boundary surfaces of computational nodes, the coupled net current equations with boundary conditions can be written as following matrix form:

$$[M_{gu}]\cdot \underline{J}_{gu}=\underline{S}_{gu} \qquad (5)$$

The above matrix equation is solved by source iteration method.

The function of the code is similar to that of SIMULATE-3. A number of well established benchmark problems[7,8,9] have been employed to validate NNGFM code. The detailed description about the validation tests is referenced to Hu and Zhao[5]. The preliminary reactor physics design of NHR-200 with NNGFM code has also passed the license assessment of Chinese nuclear safety administration.

<u>ORFM</u> is developed by INET[10,11,12] recently for performing Optimization of Reactor Fuel Management calculation, including load pattern reshuffling to optimize the cycle length, power peaking factor. The development of the code package is partially supported by EDF (Electricité de France) of France. This package adopts following algorithms, namely SA(Simulated Annealing), GA (Genetic Algorithm) and SIA (Statistic Inductive Algorithm) for optimization calculations. Although NNGFM code currently is employed as the default reactor core physics design code, the interfaces for other reactor core physics design codes, such as SIMULATE-3, COCCINELLE and MapNet3 based on ANN (Artificial Neural Networks) scheme developed by EDF, are also prepared. Both SA and GA algorithms are widely used for in-core fuel management optimizations. The main development in ORFM code related to SA and GA applications is the combination of the algorithms with some knowledge-based local search methods, such as crossover and an adaptive method similar to "ant pheromone evaporation" scheme [13], in order to avoid early trap due to local optima and to enhance the capability of the routines with SA and GA algorithms for searching global optimized solutions. The SIA algorithm used in ORFM code established originally at INET of Tsinghua University. The main idea of the SIA algorithm is to preserve the statistic distribution of selected objective functions in processing population evolution. This new algorithms is superior to traditional GA algorithm mainly in searching better solutions from generation to generation while the best solutions will never evaporate or degenerate. The SIA algorithm can be regarded as a kind of Adaptative Memory Programming (AMP)[14] introduced by Taillard[15] with a

unified version of meta-heuristics. The main procedure of implementing the SIA algorithm can be sketched as follows

1) Memory in SIA is initiated by N1 selected solutions in arbitrary way and the initial statistical distribution of the objective function referring to each initial solution are generated and stored in solution search process.
2) Provisional solutions are randomly generated and modified in such a way of preserving the statistic distributions generated in step 1). This process may be referred to the reproduction of the new generation of population in GA algorithm.
3) Improvement procedure similar to the mutation in GA algorithm based on local search is actually embedded in step 2) when provisional solutions are produced.
4) Memory update follows the culling strategy by keeping the best N1 solutions from the successive two generations of solutions while eliminating worst solutions.

The above steps are repeated till the prescribed criterion limiting the search number for new generation of solutions is exceeded.

Because SIA algorithm is rather efficient in performing in-core fuel management optimization, implementing it to ORFM code package makes the code be possible to perform both in-core mono-cycle and multi-cycle reloading optimizations. The SIA algorithm has been tested with EDF standard 900MWe PWR monocycle load pattern problems of Faible Fluence Généralisée (general low flux) referring to Blayais-314 and Cruas-117. The results have shown that SIA algorithm is efficient to find good solutions. Although the Ant System (AS) algorithm may be slightly faster in searching optimized solutions in small search space (select 1/8 symmetric core and fix the positions of fresh and very old fuel assemblies), SIA algorithm becomes more efficient than AS method with an enlarged searching space ranging from 1/8 core to 1/4 core. The preliminary tests with SA and GA methods in optimizing Dayabay 900MWe PWR and NHR-200 in-core load pattern reshuffling have been also quite satisfactory.

TQE/NGFM is a reactor dynamics analysis code developed at INET[9] by employing the temporal quadratic expansion nodal Green's function method.

$$\frac{1}{V_g}\frac{\partial\Phi_g}{\partial t} = \nabla\cdot D_g\nabla\Phi_g - \Sigma_g^R\Phi_g - \sum_{g'\neq g}^{G}\Sigma_{g'-g}^{s}\Phi_{g'} + (1-\beta)\chi_g\sum_{g'=1}^{G}\nu\Sigma_{g'}^{f}\Phi_{g'} + \sum_{i=1}^{I_d}\chi_{ig}\lambda_i C_i$$

$$\frac{\partial C_i}{\partial t} = \beta_i\sum_{g'}^{G}\nu\Sigma_{g'}^{f}\Phi_g - \lambda_i C_i\,, \quad g=1,\dots,G; \quad i=1,\dots,I_d \tag{6}$$

By expanding time dependent neutron flux with a quadratic function at time n, the time derivative of neutron flux and the concentration of fission precursors can be expressed as the arithmetic function of the neutron fluxes at two different time points within the current time step. Then the concentration of fission precursors can be replaced by the time dependent neutron fluxes and the time-and-space dependent neutron dynamic equation can be converted into a steady state-like multi-group neutron diffusion equation at time n+1 as follows:

$$-\nabla\cdot D_g^{n+1}(r)\nabla\Phi_g^{n+1}(r) + \Sigma_{R,g}^{n+1}\Phi_g^{n+1}(r) = S_g^{n+1}(r), \quad g=1,\dots,G \tag{7}$$

The Green's function method described previously for NNGFM code is used to derive an matrix equation referring to neutron flux distribution at time n+1 from the Eq. (7) by assuming continuity of partial neutron flows in space. Then neutron flux at time n+1 can be obtained by iterative method from the deduced matrix Eq. (7).

The validation test calculations carried out so far has provided acceptable results although more validations are still needed. The code is planned to couple with some thermal-hydraulic system codes, such as Relap5/MOD3.2, RETRAN-02 and ATHLET etc., to form a user friendly engineering plant analyzer for NHR-200. The code development work is still underway.

4. CONCLUSIONS

The main design features of NHR-200 nuclear heating reactor developed by INET of Tsinghua University have been summarized in a brief form in this paper. The reactor physics aspects of NHR-200 core and the design codes have been addressed. The algorithms and the functions of the reactor physics design codes for NHR-200 are presented and their qualifications as design tools are commented.

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