Research Journal of Applied Sciences, Engineering and Technology 6(11): 2019-2026, 2013
DOI:10.19026/rjaset.6.3818
ISSN: 2040-7459; e-ISSN: 2040-7467

Submitted: November 24, 2012 Accepted: January 17, 2013 Published: July 25, 2013

Research Article

Three Dimensional Numerical Simulation on Nuclear Reactor Interior Flow and Temperature Field of a 1000MW Unit

[1, 2]Jian-Quan Liu, [1]Bao-Min Sun, [2]Zhu-San Li, [2]De-Liang Zhang, [2]Ling-Yun Zhou and [1]Tao Bai
[1]Key Laboratory of Condition Monitoring and Control for Power Plant Equipment, North China Electric Power University, Beijing, China
[2]Fujian Ningde Nuclear Power Company Limited, Fujian, China

Abstract: The Flow and Temperature Field of a 1000 MW Unit Nuclear reactor operation process was numerically studied with standard k-ε model. The conditions of reactor normal, circuit and inlet failure were studied. The different reactor flow characteristics between normal operating and two kinds of failure cases were compared. The numerical simulation and experiment result as follows: In normal operation condition, coolant of three circuits is distributed in symmetry along circumference of the reactor inlet ring. The coolant flow surrounding and center is relatively large before the reactor core inlet. At upper grating plate region, the coolant flow is symmetry distributed along the grating plate. In case that a circuit fails suddenly and the coolant totally loses, the distribution of water flow for operation circuits is unevenly increased rapidly. In case three circular passages at inlet of the reactor is unevenly blocked by sundries, when a circuit fails suddenly and coolant is totally lost, asymmetry of water flow distribution for three circuits will increase further, which may accelerate the degree and probability of damage to reactor core.

Keywords: Circuit failure, inlet failure, flow and temperature field, numerical simulation, standard k-ε model

INTRODUCTION

1000WM pressurized water reactor studied in the paper is mainly composed of rector pressure vessel, components of reactor core, components of reactor internals, rector pressure vessel head and control rod drive mechanism. Among them, the components of reactor core produce the heat energy, which mainly consists of control rod assembly, burnable poison assembly, neutron source assembly and thimble plug assembly. The reactor makes use of the pressurized light water as the moderator and coolant, the reactor pressure vessel is located at the center of reactor containment on the nuclear island, which can contain the reactor core, at the same time, which is primary-to-secondary barrier and the heat can be out flowed.

The detailed change condition of hot-state flow field inside the reactor can not be measured in operation test and the workload is huge, the numerical simulation can reflect the change process of flow field inside the reactor in details and it has already been extensively used in industrial research (Sohankar and Davidson, 2001; Islam *et al*., 1998; Patankar, 1980; Wang, 2004). Which is difficult to be achieved by actual test. Meanwhile, relevant document also shows that properties of numerical simulation are basically similar to practical situation (Oka, 2000; Jeong *et al*., 2007; Lee and Jang, 1997) in the process of fluids flow in the reactor. In this study, we perform a comparison between results of numerical calculation and situations of operation test, they possess the high-degree consistency in trend and the calculation in this study is indicated to be with much accuracy through comparison.

LITRATURE EREVIEW

The 1000MW unit to study in this study is made up of 3 separate cooling systems, i.e. the Reactor Coolant System, refers to Fig. 1, the Secondary and the third circuit. The Reactor Coolant System (inside the Containment) consists of 3 Cooling "Loops" connected to the Reactor, each containing a Reactor Coolant Pump and a Steam Generator. The Reactor heats the water that passes through the fuel assemblies from a temperature of about 290°C to a temperature of about 320°C. Boiling is not allowed to occur. The pressure is maintained by the Pressurizer connected to the loop 1 at 15.5 Mpa. Flow process for one-circuit coolant is as follows, the main pump drives one-circuit coolant to force circulatory flow and brings the heat produced through fuel fission outside the reactor to steam generate, the heat is transferred to two-circuit feed-water through steam generator. The coolant cools the

Corresponding Author: Jian-Quan Liu, Key Laboratory of Condition Monitoring and Control for Power Plant Equipment, North China Electric Power University, Beijing, China

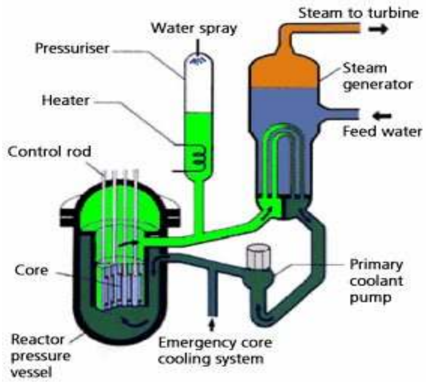


Fig. 1: 1000MW unit pressurized water reactor and primary coolant system

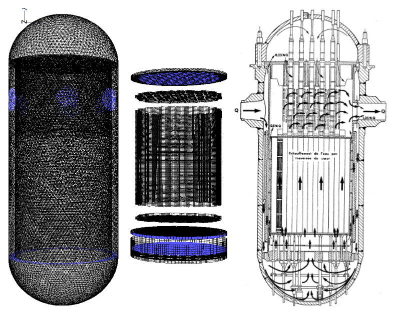


Fig. 2: Reactor calculation domain mesh and flow condition

reactor core in the process of leading heat out of the reactor core so as to prevent the fuel rod from burning, The nuclear reactor being studied in this study belong to one-circuit equipment.

Aiming at design principle of internals in 1000WM pressurized water reactor, no discussion is performed in the study. And the study mainly aims at calculations for automatic adjusting ability of system under circuit failure and partial stoppage conditions, among circular passages during operation process of the reactor. So it can provide the theoretical basis for emergency trip as well as judgment for reason of failure under accident condition of the reactor in power plant.

The geometric model with full dimension is adopted for numerical simulation of pressure vessel and reactor core of the reactor. 3D solid modeling is conducted for reactor core component and the whole flow passage. Unstructured tetrahedral mesh is adopted for the calculation area (Baglietto and Ninokata, 2005; Heina and Mantlik, 1988). As for numerical simulation of reactor pressure vessel and rector core, in consideration of relatively rapid change of flow field in the core area, the division for mesh is relatively fine and detailed in order to guarantee that water flow passage of the grating plate at the core inlet, core heat exchange areas and water flow passage of the grating plate at the core outlet are in different meshes to avoid the influence of pseudo-diffusion. Because of the large amount of calculation for three-dimensional full dimension of nuclear reactor and according to the calculation in this study, the area from inlet pipeline of the reactor to inlet plate of reactor core is defined as inlet passage. The area from inlet plate of reactor core to its outlet plate is defined as central passage and the area from outlet plate of reactor core to outlet pipeline of the reactor is defined as outlet passage. The upper grating plate of the reactor core is selected as an object of research to gradually enable the mesh to become dense uniformly and calculate it. In comparison with the previous calculation result, the calculated convergence result is less than 10^{-4} and difference for results of 2 times of calculation is within 5%. Total number of the divided mesh for reactor is approximately 2.98×10^6. The mesh arrangement and flow condition are shown in Fig. 2.

The eddy viscosity turbulence model is selected for numerical study in this study. Three-dimensional calculation for stable state is adopted for the numerical simulation. And for simulation of turbulence flow in liquid phase, standard k-ε model for simulation is applied for calculation of turbulence. k-ε model is the most commonly used turbulence model, which is extensively applied in numerical simulation of boundary layer type flow and flow inside the tube (In *et al*., 2003; Wang *et al*., 2005). Its general formula for equation as follows:

$$\frac{\partial(\rho k)}{\partial t}+\frac{\partial(\rho k u_i)}{\partial x_i}=\frac{\partial}{\partial x_j}\left[\left(u+\frac{u_t}{\sigma_k}\right)\frac{\partial k}{\partial x_j}\right]+G_k+G_b-\rho\varepsilon-Y_M+S_k \tag{1}$$

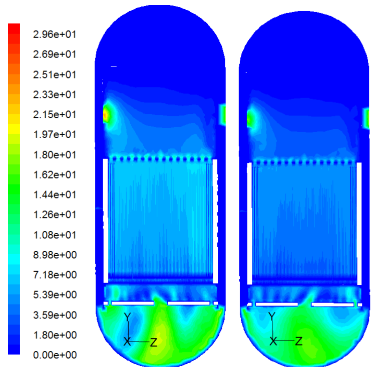
$$\frac{\partial(\rho\varepsilon)}{\partial t}+\frac{\partial(\rho\varepsilon u_i)}{\partial x_i}=\frac{\partial}{\partial x_j}\left[\left(u+\frac{u_t}{\sigma_\varepsilon}\right)\frac{\partial\varepsilon}{\partial x_j}\right]+C_{1\varepsilon}\frac{\varepsilon}{k}\left(G_k+C_{3\varepsilon}G_b\right)-C_{2\varepsilon}\rho\frac{\varepsilon^2}{k}+S_\varepsilon \tag{2}$$

In Eq. (1) and (2):

$$G_k=u_t\left(\frac{\partial u_i}{\partial x_j}+\frac{\partial u_j}{\partial x_i}\right)\frac{\partial u_i}{\partial x_j}\quad G_b=\beta g_i\frac{u_t}{\mathrm{Pr}_t}\frac{\partial T}{\partial x_i}\quad \beta=-\frac{1}{\rho}\frac{\partial\rho}{\partial T}$$

$$Y_M=2\rho\varepsilon M_t^2\quad M_t=\sqrt{k/a^2}\quad \alpha=\sqrt{\gamma RT}$$

It is assumed that the movement for fluids has been sufficiently developed. Parameters refer to reference literature (Yakhot *et al*., 1992; Lien and Leschziner, 1994), parameters refer to reference literature (Wang *et al*., 2005; Li *et al*., 2009). The coupled calculations are conducted for fluids on the surface of core component and inside the passage. In addition, the discrete equation is established by adopting the finite volume method and discrete format of each control formula is employed for second-stage windward format. SIMPLEC algorithm is adopted for coupled mode of velocity pressure. Value for inlet is taken according to pipeline conditions, while the value for outlet is taken according to fully developed conditions. In calculation of this study, the influence of leaking flow on core flow is omitted and the selection of other parameters are in strict consistency with practical operation conditions.



(a) (b)

Fig. 3: Reactor longitudinal section flow field (unit: m/s: (a): Ordinary condition; (b): circuit failure)

RESULTS AND DISCUSSION

So as to achieve a convenient comparison, Z negative is defined as no.1 circuit inlet, Y positive is defined as height respectively. The middle position of the horizontal upon the core inlet plate 900 mm is X-axis 0 point.

Analysis on characteristics of arrangement for full passage flow field of the reactor: Total amount of coolant for the reactor will be remained as 3×23,790 m^3. Three circuits are evenly arranged. The pressure value for inlet coolant is 15.5Mpa and temperature of inlet coolant is 292.7°C. It is defined as rated condition (follows referred to as rated condition). In numerical simulation for characteristics of flow field of the overall reactor, reactor longitudinal section flow field is shown in Fig. 3. It is found that when the coolant fluid flows through obstacle, a series of vortexes will be produced behind it Oldekop *et al*. (1982), Krauss and Meyer (1998) and Manoran and Gomez (1995). In the generated swirls, there are mainly Transverse Vortex (TV) and Longitudinal Vortex (LV), as indicated in Fig. 3. The coolant enters into circular passage of interlayer in the reactor through the inlet tube, because thermal shield assembly and male pin are installed among different ring circuits inside the circular passage of interlayer, thermal shield assembly and male pin produce much action on fluid. Meanwhile, because the water flow is in horizontal direction, transverse vortexes inside the circular passage are not clear with comparison of longitudinal vortex. Inlet water flow of the three circuits along internal circular passage is provided with good uniformity.

There is a spherical area in the lower part of reactor. The coolant will produce a large number of transverse vortex and longitudinal vortex after entering into the spherical area from circular passage, longitudinal vortex in central area and surrounding area is quite clear. The overall flow field is comparatively stable, but water flow in former area of core inlet is not

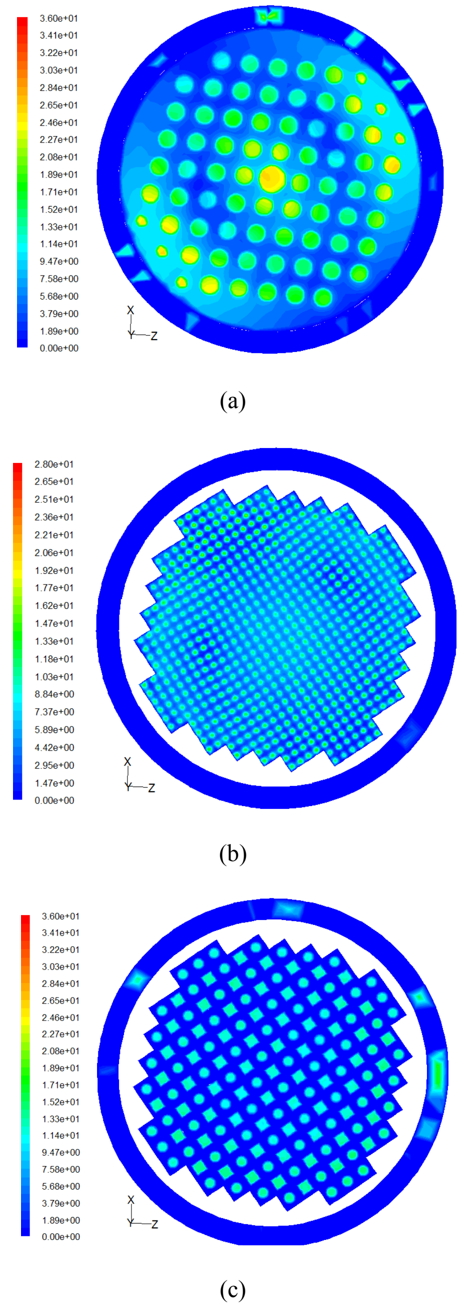


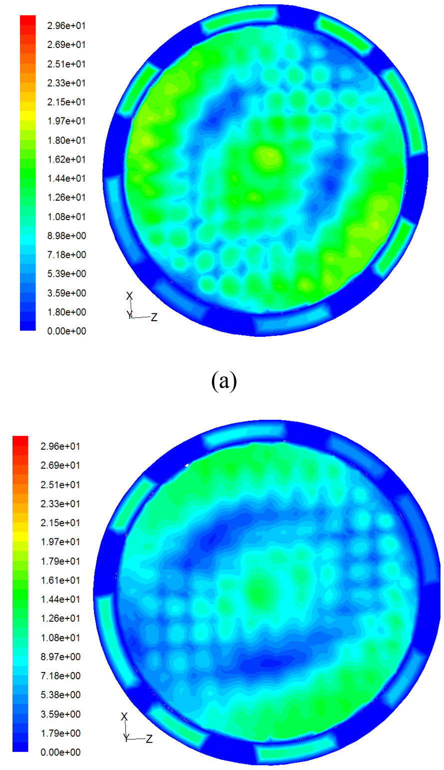
Fig. 4: Flow field characteristics of distribution for reactor core (unit: m/s); (a) Core inlet plate; (b) Lower grating plate; (c) Upper grating plate

evenly distributed along the horizontal cross-section. After the coolant flows through core inlet plate, water flow entering into the core will be redistributed along horizontal cross-section due to redistributing function of water hole in core inlet plate. The flow field undergoes a greater change before entering into lower grating plate of the core. In case of different flows, there is certain difference in distribution of water flow entering into the lower grating plate of core. Under different coolant flows, distributing characteristics along horizontal cross-section at the upper grating plate of the core tends are uniform with comparability of that along upper grating plate of the core.

Analysis on characteristics of flow field distribution for reactor core: Reactor will be remained to be operation under rated condition. The inlet water flow at lower grating plate of the core is shown in Fig. 4a. In addition, there are 157×4 inlet holes in total for lower grating plate of core, the inlet holes regularly distribute along lower grating plate. All coolant entering through the inlet holes at lower grating plate of the core. It can be observed from calculation results in the Figure. The water flow is not evenly distributed at cross-section of lower grating plate of the core. The water flow velocity is much rapid in surrounding and central areas. Its actual flow is on the larger area. With a comparison to Fig. 4b and c, it can be discovered that the water flow entering into the core is not stable in the lower area of the core, Due to coordinated action of fuel component, internal components and other assemblies inside the reactor, the coolant tends to be uniform in distribution along horizontal cross-section at upper grating plate area of the core.

The included angle between inlet and outlet of every circuit (total three circuits in the rector) is 50º. Its inlet and outlet are arranged in clockwise with proper order. The circular passage between reactor pressure vessel and core barrel belongs to the passage of coolant. The three circuits are arranged along the uniform angle of circumference and the thermal shield assembly is designed as four assemblies, which is corresponding to the core internal structure. Therefore, the thermal shield assembly and the three circuits are designed to unsymmetrical structure. The calculation results show that asymmetry about three circuit water flows is poor and there is very little influence to total flow field under normal condition. Water flow of the reactor pressure vessel inlet and outlet is shown as follows, coolant was put into reactor core in the lower core plate and water flow is evenly distributed along the reactor outlet through the mixed influence of core internals and upper core plate. However, it can be seen through analysis. The isopotential is not uniformly distributed at the cross-section. It is intense in areas near outlet and circumference, actual resistance for water flow at cross-section of the outlet is asymmetry.

Analysis on distribution characteristics for flow field of reactor in case of circuit failure: Under the circumstance that a circuit fails suddenly and the coolant completely loses, total amount of reactor coolant will be 2×23790 m^3. The pressure value of inlet coolant is 15.5Mpa and the temperature of inlet coolant is 292.7°C. Three and two circuit condition Inlet water flow before core lower grating plate show in Fig. 5, inlet water flow of core lower grating plate is as shown in Fig. 6b and the outlet water flow is as shown in Fig. 6c, while the Fig. 6a shows the water flow distribution before water flow enters into fuel cluster. And it can be observed from the calculation results in the Figure that the coolant is unevenly strengthened at inlet of core lower grating plate and water flow in areas of circumference and center is in relatively large velocity.



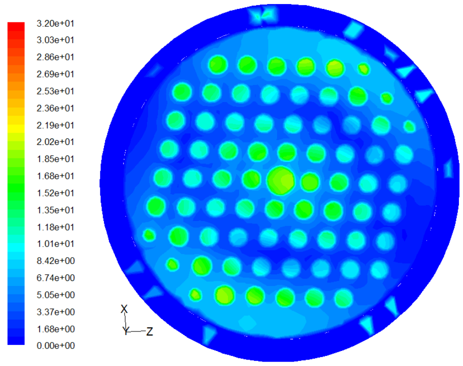
(a)

(b)

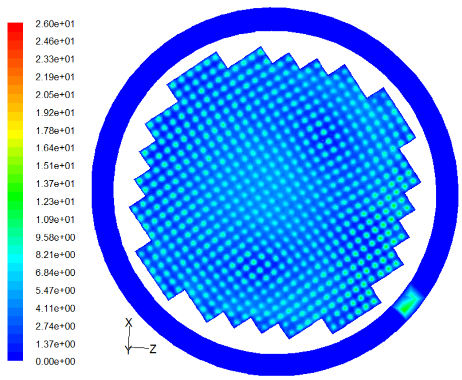
Fig. 5: Flow field characteristics of reactor in case of circuit failure (unit: m/s); (a) Three circuit condition; (b) Two circuit condition

The water flow of three circuits has increased unevenly in distribution. Water flow at inlet of core lower grating plate possesses much more clearly thermal shield assembly in comparison with that at inlet of core upper grating plate. Although the core inlet plate and inlet of core lower grating plate are provided with a certain regulating effect on thermal shield assembly of water flow, they cannot eliminate the phenomenon of uneven flow caused by failure of circuit and there is great asymmetry in core cooling.

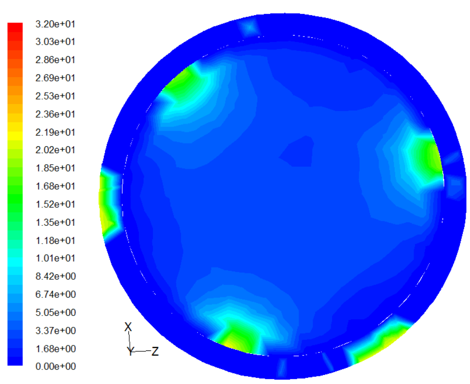
Temperature field of reactor in case of circuit failure can be seen in Fig. 8, as show in It can be observed with a comparison to Fig. 3, the coolant is still provided with great uneven distribution of flow before entering into the fuel rod area. Therefore, under condition of circuit failure, the core coolant will come out the phenomenon of uneven flow. Under this circumstance, the insufficient cooling for partial places of core will be brought and the deteriorated heat transfer may produce in partial areas. Under this



(a)



(b)

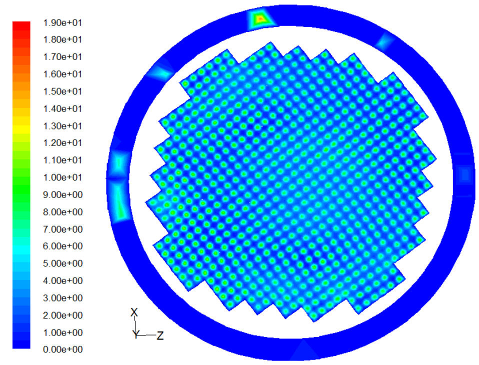


(c)

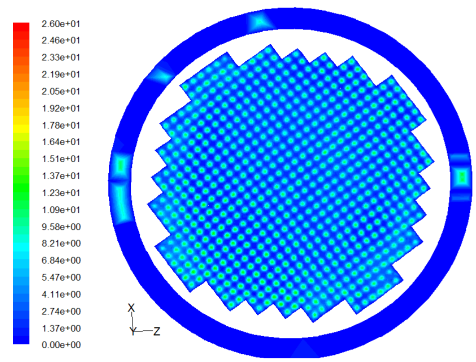
Fig. 6: Flow field characteristics of reactor in case of circuit failure (unit: m/s); (a) Core inlet plate; (b) Lower grating plate; (c) Inlet and outlet section

condition, the reactor will be shut down in emergency in order to avoid the accident of core damage due to insufficient cooling of core at partial places.

It is shown by results of numerical simulation that the thermal shield assembly is designed between pressure vessel and core barrel in the middle of circular passage. Under the normal operation condition, the water flow by three circuits is uniformly distributed along the circumference of the core. However, it is not evenly distributed along the circular passage and water flow tends to be on the small side for lower part of the blocked flow area.



(a)



(b)

Fig. 7: Flow field characteristics of reactor in case of inlet failure (unit: m/s); (a) Circuit ordinary condition; (b) Circuit failure condition

Analysis on distribution characteristics for flow field of reactor in case of inlet failure: Total amount of coolant for the reactor is remained as 3×23790 m^3 and three circuits are evenly distributed. The inlet coolant pressure value is 15.5Mpa. The coolant temperature of inlet is 292.7°C. Situations of passages of three circuits being blocked by sundries will be simulated. The calculation result refers to Fig. 7. It is shown in the calculation result that the coolant is evenly increased at outlet of circular passage and water flow at core area is relatively stable. The uneven distribution of water flow for three circuits is weakened and it is attributed to symmetric structure of four pieces of thermal shield assembly and three circuits, which also exerts an impact on uniformity at outlet of circular passage for coolant. Three circular passages at the inlet are evenly blocked along the circumference in its middle. Therefore, three circuits form into separated passages mutually. Since there is a coolant pump for every three circuits for circulatory drive of the fluid, flow distribution uniformity of the coolant at outlet of circular passage increases.

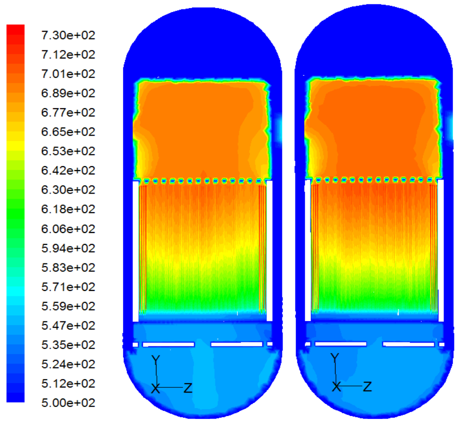


Fig. 8: Temperature field of reactor in case of circuit failure (unit: K); (a) Three circuit; (b) Two circuit

If three circular passages are blocked by sundries along the circumference and are not symmetric (its calculation results are shown in Fig. 7, after the coolant enters into the fuel rod, the uniformity for distribution of flow has been strengthened with comparison to that under original condition. Under circuit failure, seriously uneven flow will occur to the core coolant, this will intensify the damage to the core. Therefore, the inlet circular passage at reactor pressure vessel will be remained for design of condition resistance coefficient to ensure the security under accident condition.

In case that a circuit fails suddenly and coolant drive totally loses distribution of water flow for three circuits is unevenly increased and regulating effect of lower plate of core inlet and lower grating plate on water flow will not eliminate uneven flow due to circuit failure. Before the coolant enters into the area of fuel rod, it possesses very seriously uneven flow. Under the circumstance that circular passages for the three circuits are unevenly blocked by sundries, the uneven distribution of coolant will increase further, which will accelerate the degree and probability of damage to reactor core.

Analysis on distribution characteristics for temperature field of reactor: During the well-balanced hot functional operation, the fuel module heat up the refrigerant in core. The temperature of the refrigerant goes high with the height of water lever ups. As shown in Fig. 8a, we can find that, the lower part of the reactor core is not symmetrical, with the function of fuel module and the reactor component. The temperature distributing goes more and more symmetrical. Compare to the temperature circulation and flow circulation distribution picture, we can find some pertinency between these two conditions. In high temperature area the flow shows more tempestuousness thus accelerate the refrigerant cooling the fuel module. That cause the temperature of this area lower. Figure 8b shows the distributing of temperature while one of the refrigerant pumps is lost. compare to Fig. 8a we can find that the temperature of the refrigerant goes up faster, the high temperature area goes down . the exit flow temperature goes higher, some of the fuel area's temperature goes up to 695K, the temperature of the fuel stick approach to unstable temperature. In this status, the nuclear reactor needs to fall power to shut down the reactor.

ACTUAL EXPERIMENT RESULT

When the reactor is put into operation, two kinds of arranging methods about fuel around enrichment and fuel center enrichment is compared. Two kinds of fuel enrichment, the reactor can operate stable. In the surrounding fuel-rich case, the surrounding coolant flow is relatively large, so cooling effect is good. Under high-load condition, the flow along the vertical density difference increases and accelerates the flow rate. LingAo Nuclear Power Station No.1 and No.2 unit reactor are designed to load surrounding fuels enrichment, the reactor can operate stable. The numerical results are consistent with the actual operation flow trends. For center fuel enrichment case, the reactor axial central region has a greater coolant flow and most of the control rod in center area, reactor power can be in very good control; LingAo Phase II unit reactor are used the central fuel-rich loading mode, compared to the reactor internal components displaying the measured data, numerical simulation results and the actual operation condition of the reactor show a high degree of consistency. The numerical results are consistent with the experimental data, so that the numerical method can correctly predict Reactor outlet flow. Calculation and Measured contrast on different Coolant pump arranging show the same trace.

When the reactor is arranged to hot-state operation test, the test purpose is to check a partial loss of forced reactor coolant flow accident. This series of flow test is carried out during Ningde nuclear power plant unit 1 reactor. Different coolant pumps failure condition is arranged in this test. Calculation and experimental data contrast two reactor coolant pumps failure condition as shown in Table 1. In the test, No.1 loop reactor outlet pressure is calculated accord to design value and takes into account operation coolant pump suction, failure loop (i.e., No.2 and No.3 loop) reactor outlet pressure is accord to design value by oneself. Calculation and experimental data of reactor outlet flow on different Coolant pump arranging are Compared, deviation of the numerical results is less than 12%, failure conditions of other coolant pumps also show the same flow trace.

The circulation temperature to calculate was based on designed functional status, the average heat flow was set to 62W/m^2. Keep the refrigerant gross is the design capacity, the entrance pressure is 15.5Mpa, the temperature is 292.7°C, the exit temperature and the design temperature warps is under 15%. Which show

Table 1: Calculation and experimental data contrast of two reactor coolant pumps failure condition

Coolant circle	Coolant pump arrangement	Calculation flow	Experimental flow
No.1	No.2 and No.3 coolant pump failure	16.86 m/s	17.41 m/s
No.2	No.2 and No.3 coolant pump failure	4.18 m/s	4.16 m/s
No.3	No.2 and No.3 coolant pump failure	4.17m/s	4.25 m/s

that the calculation the same trace. to actual condition and show the same circulation temperature characteristic in the reactor.

CONCLUSION

- Results of analog calculation coincide closely with the test results qualitatively, which shows that the numerical calculation for internal flow field of pressurized water reactor vessel by using CFD is feasible.
- Circular passage is designed between pressure vessel and core barrel, under normal operation condition, coolant of three circuits is distributed in symmetry along circumference of the ring and its distribution possesses certain asymmetry.
- In case that a circuit fails suddenly and the coolant totally loses, the distribution of water flow for operation circuits is unevenly increased and regulating effect of lower plate of core inlet and lower grating plate on water flow will not eliminate uneven flow due to circuit failure.
- In case those three circular passages at inlet are unevenly blocked by sundries, flow of outlet coolant for circular passages is distributed asymmetrically along circumference. Three circular passages are evenly blocked by sundries. In case that a circuit fails suddenly and coolant totally loses, asymmetry of flow water distribution for three circuits will increase further, which may accelerate the degree and probability of damage to reactor core.

ACKNOWLEDGMENT

This study was supported in part by Center high school basis scientific research operation expert Foundation Project 09ZG02 and in part by the Fundamental Research Funds for the Central Universities.

REFERENCES

Baglietto, E. and H.A. Ninokata, 2005. Turbulence model study for simulating flow inside tight lattice rod bundles [J]. Nuclear Eng. Design, 235: 773-784.

Heina, J. and F. Mantlik, 1988. The structure of turbulent flow in finite rod bundles [C]. Proceeding of 1st World Conference on Experimental Heat Transfer, Fluid Mechanics and Thermo Dynamics. Elsevier, Amsterdam, pp: 1712-1719.

In, W.K., D.S. Oh and T.H. Chun, 2003. Simulation of turbulent flow in rod bundles using eddy viscosity models and the Reynolds stress model [C]. Proceedings of the 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10). Seoul, Korea.

Islam, M.S., R. Hino, K. Haga, M. Monde and Y. Sudo, 1998. Experimental study on heat transfer augmentation for high heat flux removal in rib-roughened narrow channels [J]. Nuclear Sci. Technol., 35(9): 671-678.

Jeong, H.Y., K.S. Ha, Y.M. Kwon, Y.B. Lee and D. Hahn, 2007. A dominant geomet-rical parameter affecting the turbulent mixing rate in rod bundles [J]. Heat Mass Transfer, 50: 908-918.

Krauss, T. and L. Meyer, 1998. Experimental investigation of turbulent transport of momentum and energy in a heated rod bundle [J]. Nuclear Eng. Design, 18: 185-206.

Lee, K.B. and H.C. Jang, 1997. A numerical prediction on the turbu-lent flow in closely spaced bare rod arrarys by a nonlinear k-model [J]. Nuclear Eng. Design, 172: 351-357.

Li, Y., W.X. Zhou, J.G. Zhang and D.Z. Wang, 2009. Numerical simulation of three dimensional flow through full passage and performance prediction of nuclear reactor coolant pump [J]. Atomic Energy Sci. Technol., 43(10): 898-902.

Lien, F.S. and A. Leschziner, 1994. Assessment of turbulent transport models including nonlinear RNG eddy viscosity formulation and second moment closure for flow over backward facing step [J]. Comput. Fluids, 23(8): 983-1004.

Manoran, J.V.S. and M.O. Gomez, 1995. Alternating direction implicit method with exponential upwinding. Comput. Math. Appl., 30(11): 47-58.

Oka, Y.K., 2000. Design concept of once-through cycle super-critical pressure light coolant cooled reactors [C]. Proceeding of SCR-2000. Tokyo, Nov. 6-8, pp: 1-22.

Oldekop, W., H.D. Berger and W. Zeggel, 1982. General features of advanced pressurized water reactors with improved fuel utilization [J]. Nuclear Technol., 59: 212-227.

Patankar, S.V., 1980. Numerical Heat Transfer and Fluid Flow [M]. Hemisphere, New York, pp: 140.

Sohankar, A. and L. Davidson, 2001. Effect of inclined vortex gen erators on heat transfer enhancement in a three-dimensional channel [J]. Numer. Heat Tr. A-Appl., 39(5): 433-448.

Wang, F.J., 2004. Analysis of Computational Fluid Dynamics-principle and Application of CFD [M]. Tsinghua University Press, Beijing, pp: 24-158.

Wang, J.H., H.L. Bo, S.Y. Jiang and W.X. Zheng, 2005. Numerical simulation on flow field of swimming pool reactor [J]. Atom. Energy Sci. Technol., 39(4): 322-325.

Yakhot, V., S.A. Orsazg, S. Thangam, T.B. Gatski and C.G. Speziale, 1992. Development of turbulence model for shear flows by a double expansion technique [J]. Phys. Fluids A-Fluid, 4(7): 1510-1520.