

Research Article

Sensitivity Evaluation of AP1000 Nuclear Power Plant Best Estimation Model

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The best estimation process of AP1000 Nuclear Power Plant (NPP) requires proper selections of parameters and models so as to obtain the most accurate results compared with the actual design parameters. Therefore, it is necessary to identify and evaluate the influences of these parameters and modeling approaches quantitatively and qualitatively. Based on the best estimate thermal-hydraulic system code RELAP5/MOD3.2, sensitivity analysis has been performed on core partition methods, parameters, and model selections in AP1000 Nuclear Power Plant, like the core channel number, pressurizer node number, feedwater temperature, and so forth. The results show that core channel number, core channel node number, and the pressurizer node number have apparent influences on the coolant temperature variation and pressure drop through the reactor. The feedwater temperature is a sensitive factor to the Steam Generator (SG) outlet temperature and the Steam Generator outlet pressure. In addition, the cross-flow model nearly has no effects on the coolant temperature variation and pressure drop in the reactor, in both the steady state and the loss of power transient. Furthermore, some fittest parameters with which the most accurate results could be obtained have been put forward for the nuclear system simulation.

1. Introduction

Nowadays, sensitivity analysis (SA) is becoming increasingly widespread in many fields of engineering and sciences, encompassing practically many computational modeling and process simulation activities. The sensitivity analysis is to study how the variation in the output of a model (numerical or otherwise) can be apportioned, qualitatively or quantitatively, to different sources of inputs, and how the given model depends upon the information fed into it. Furthermore, sensitivity analysis studies the relationships between information flows in and out of the model [1–4]. These definitions imply that the parameter values that characterize the boundary and initial conditions, for example, representative of a system, and the numerical structure of model correlations constitute the typical objective of SA [5, 6]. The key result from SA is the influence of input parameters upon selected output quantities and the evaluation of the relative influence of selected models, according to the definition given above.

According to Saltelli et al. (2000) [7], sensitivity analysis methods are classified into three types, local, regional, and

global. Local and regional methods evaluate the effects on the system response of small perturbations of the model input variables in the neighbourhood of some fixed, nominal values, or partial ranges of inputs variations, at low computational costs [8, 9]; the form is shown as follows:

$$\left(\frac{\partial Y}{\partial X_i} \right)_{X_j \neq X_i} (X_1^0, \dots, X_i^0, \dots, X_p^0). \quad (1)$$

The global sensitivity analysis (GSA) methods explore the whole distribution range of the model inputs and the effects of their mutual combination, which quantify the effect of all uncertain input parameters simultaneously over their ranges of variations.

Several sensitivity analysis techniques are available from the simplest of scatter plots to more sophisticated sensitivity analysis techniques, which are listed in Hoseyni et al. (2014) [10]. For example, Pearson product moment correlation coefficient is the usual linear correlation coefficient computed on (x_{ij}, y_i) , $i = 1, \dots, N$.

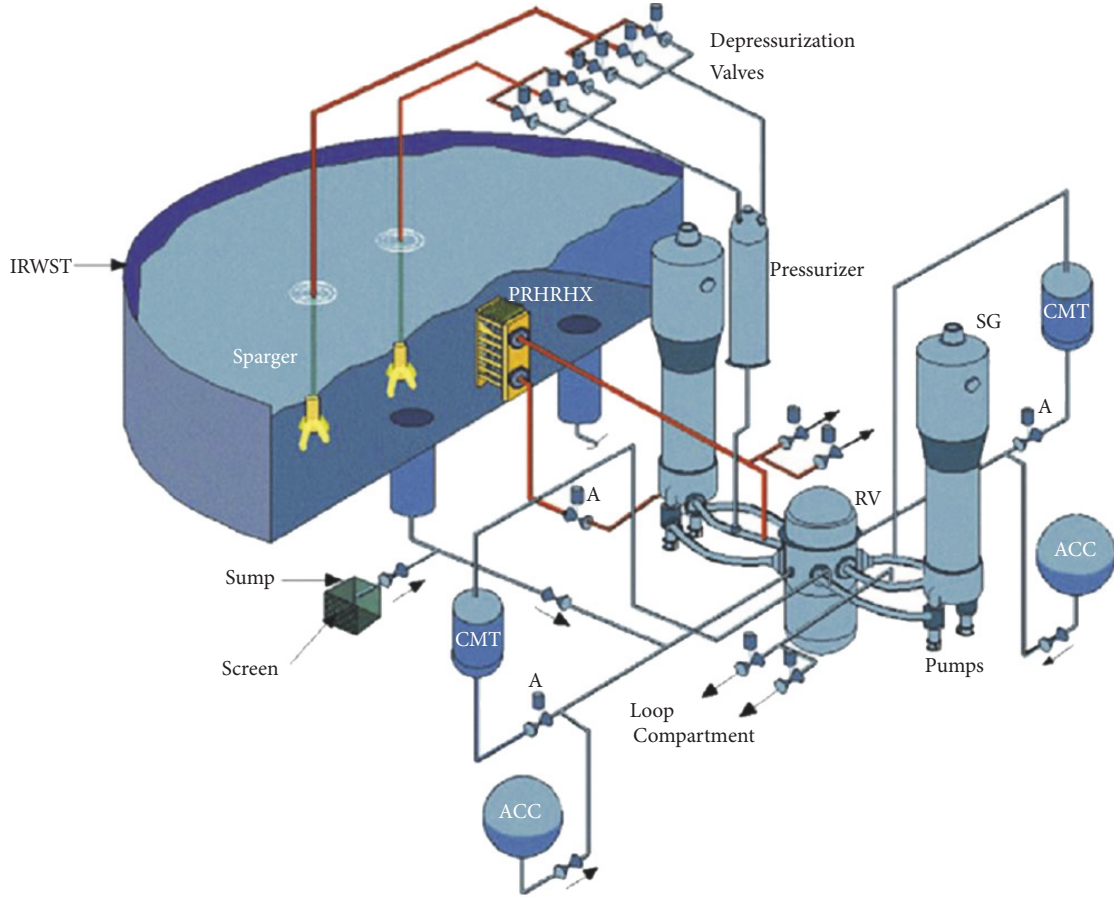


FIGURE 1: AP1000 reactor coolant system and passive core cooling system.

The Pearson (or sample) correlation coefficient (CC) between inputs x_j and output y as defined by Helton et al. (2006) [11] is

$$c(x_j, y) = \frac{\sum_{i=1}^N (x_{ij} - \bar{x}_j)(y_i - \bar{y})}{\left[\sum_{i=1}^N (x_{ij} - \bar{x}_j)^2 \right]^{1/2} \left[\sum_{i=1}^N (y_i - \bar{y})^2 \right]^{1/2}}, \quad (2)$$

where x_{ij} is the i th sample of the input x_j ,

$$\begin{aligned} \bar{x}_j &= \sum_{i=1}^N \frac{x_{ij}}{N}, \\ \bar{y} &= \sum_{i=1}^N \frac{y_i}{N}, \end{aligned} \quad (3)$$

and N is the number of samples. The Pearson correlation coefficient can also be applied to the rank transformed data and is then known as the Spearman or rank correlation coefficient (RCC).

Apart from these quantitative sensitivity analysis measures for the parameters, sensitivity importance evaluation could also be implemented qualitatively for model selection and nodal partition methods so as to make the most suitable

models and simulation processes with which the most accurate computational results could be obtained compared with the system design values [12–15].

As is known, conflicting and contradictory claims are often made about the relative models and parameter selections during the simulation processes of Nuclear Power Plants, which may cause some unpredictable influences on the code outputs. For example, when using RELAP5 code to simulate the AP1000 NPP systems, as is shown in Figure 1, the simulation results may be influenced by initial and boundary parameters such as temperature, pressure, and feedwater mass flow rate, as well as some other system conditions, for example, selected model and volume partition methods [16, 17]. Consequently, in this paper, an overall sensitivity analysis is proposed and performed to study the importance of different parameters and models upon the predicted results in the AP1000 Nuclear Power Plant and validated against available data (AP1000 Design Control Document, 2010) [18], which could assist the engineers to evaluate the effects of relative factors and improve the code simulation accuracy [19]. In addition, SA provides guidance on where to improve the state of knowledge in order to reduce the output uncertainties most effectively, especially those caused by sensitive arguments and models, so as to make the simulation programs and models more accurate [20–22].

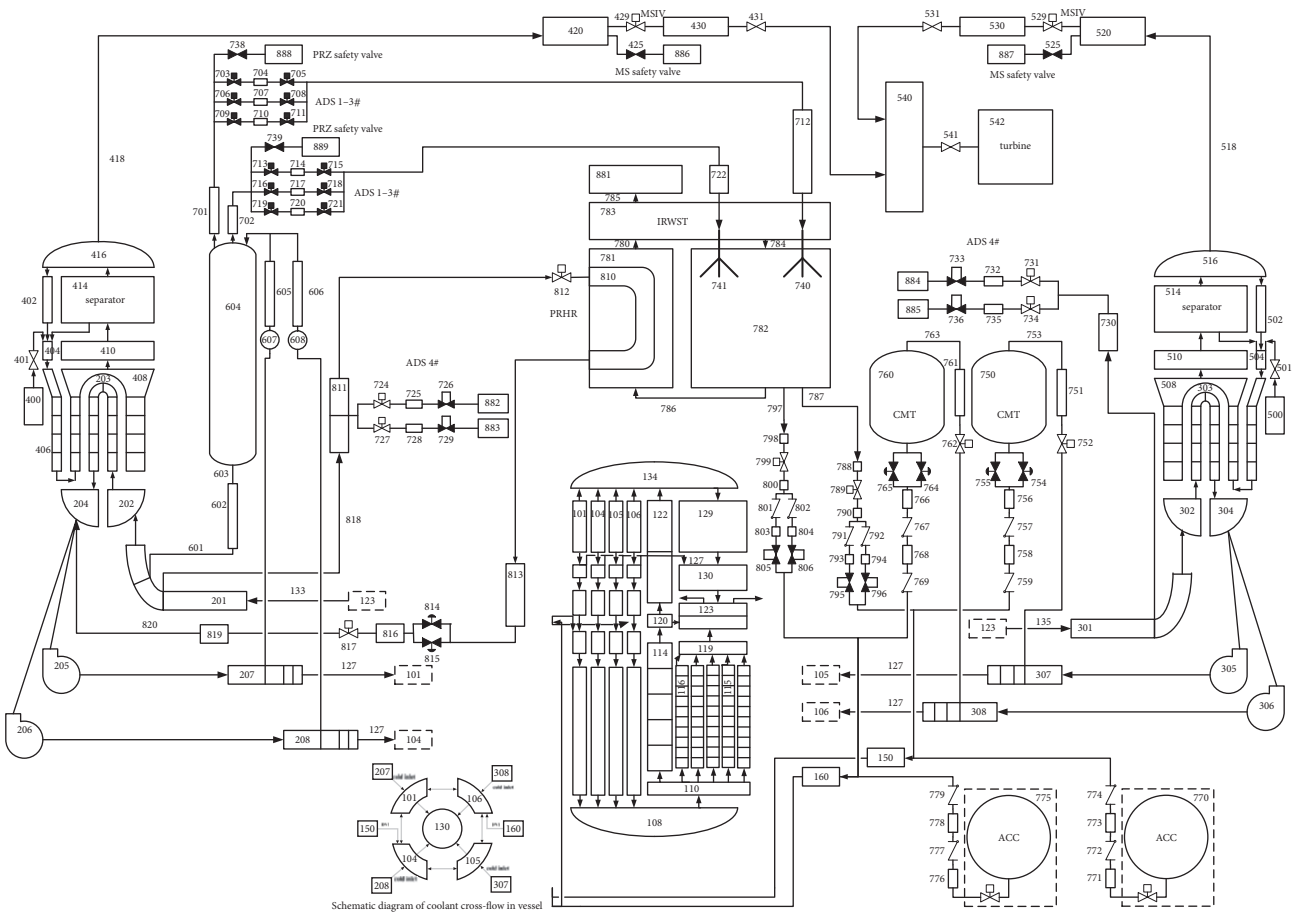


FIGURE 2: AP1000 RELAP5 model simplified scheme.

2. Establishment of AP1000 Best Estimation Model

Within the API1000 systems, the development of a RELAP5 input model started in 1999; during these years the model has been continuously improved and updated to reflect the actual plant configuration, culminating in the model documented in Ansaldo Nucleare S.p.A. (2010) [23]. Based on the best estimate thermal-hydraulic code RELAP5/MOD 3.2, a fully detailed API1000 best estimation input model has been developed and validated against available data (API1000 Design Control Document, 2010; other proprietary Westinghouse and Ansaldo Nucleare S.p.A. documents) [18, 23], as is schematically shown in Figure 2.

Figure 2 shows that the AP1000 Nuclear Power Plant reactor has two cold legs and two hot legs, which are the entrance and discharge boundaries, respectively. The four pipes (101, 104–106) simulate the annular downtake channels from different directions; control volume 108 simulates the lower plenum of the core, while the flow distribution area is simulated by volume 110. Control volume 134 simulates the upper plenum of the reactor core, the coolant flow of which comes from the annular pipes. The area between the reactor core and the upper plenum is divided into four stages of volumes; these volumes are employed to simulate the water

flow in the upper zone of the reactor [24]. The different nodal division methods of reactor core are shown in Figure 3.

As can be seen from Figure 3(a), reactor core is divided into three channels, which are simulated by pipes (112, 113, and 116) divided by ten same nodes, respectively, and pipe 114 is used to simulate the bypass channel. Besides, Figure 3(b) shows that reactor core has been divided into five channels, and five pipes (112, 113, and 115–117) divided by ten same nodes, respectively, are used to represent them, while pipe 114 simulates the bypass channel.

3. Sensitivity Evaluation Process

3.1. Sensitivity Evaluation on the Core Channel Node Number. During the simulation procedure of AP1000 NPP systems, different node division methods may have different influences on the code output; thus sensitivity analysis on core channel node number in the axial direction should be carried out during the system simulation process. According to [12, 23, 24], the core channel node number could be set at various fixed values from 5 to 50 listed in Tables 1 and 2.

3.1.1. *The Influence of Core Channel Node Number on the Coolant Temperature Variation.* According to the design

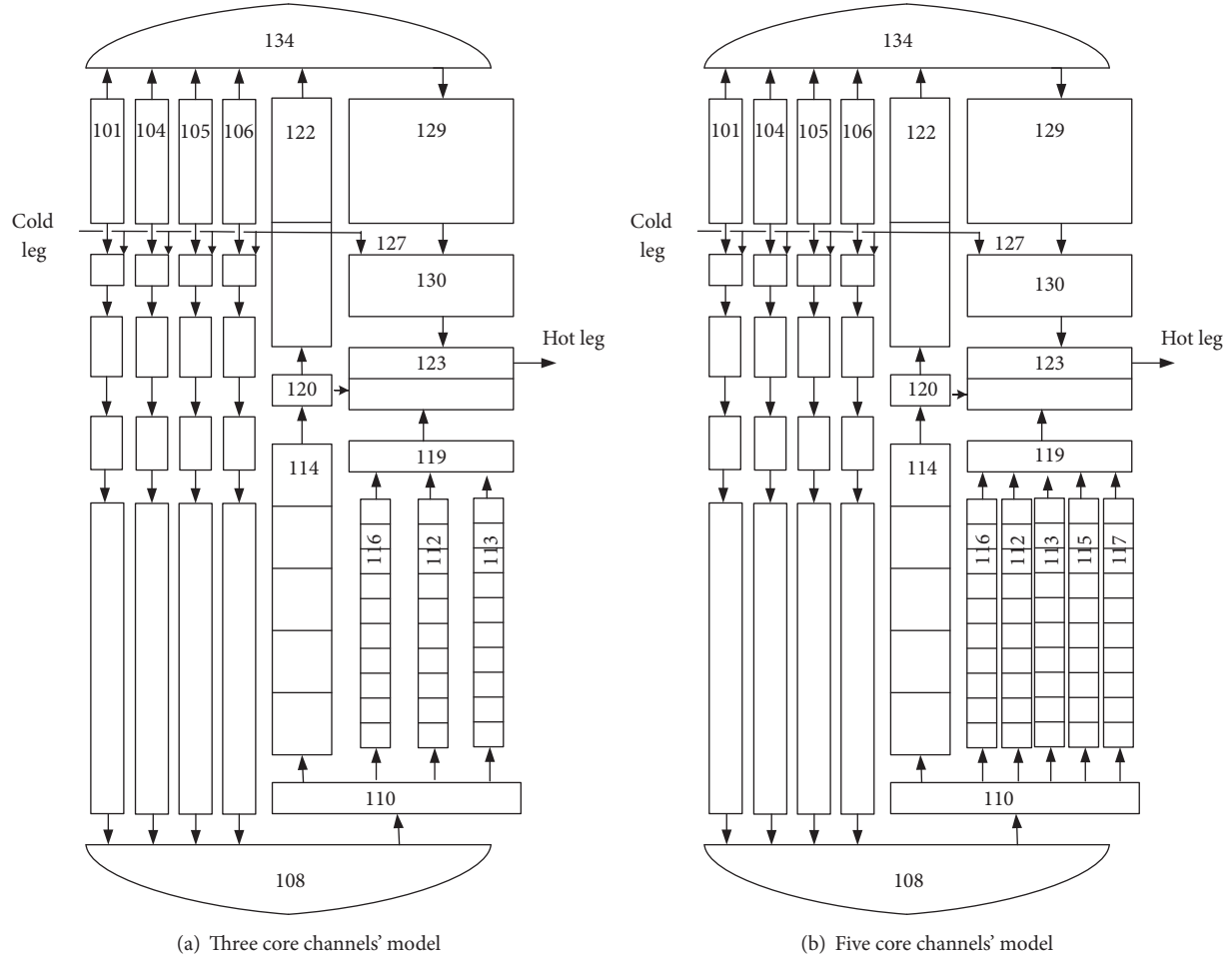


FIGURE 3: Reactor core division models of AP1000 Nuclear Power Plant.

TABLE 1: The influence of core channel node number on the coolant temperature variation.

Node number	Reactor inlet temperature (K)	Reactor outlet temperature (K)	Calculated temperature variation (K)	Designed temperature variation (K)	Fractional error
5	554.516	593.6928	39.1768	40.4	3.03%
10	554.3637	594.7499	40.3862	40.4	0.03%
15	554.2802	595.3843	41.1041	40.4	1.74%
20	554.1759	596.17845	42.00255	40.4	3.97%
30	553.95	597.64	43.69	40.4	8.14%
40	553.8906	598.1865	44.2959	40.4	9.64%
50	553.8235	598.3517	44.5282	40.4	10.22%

parameters of AP1000 Nuclear Power Plant, the reactor entrance and outlet coolant temperatures are 554.37 K and 594.77 K, respectively [25], so the coolant temperature variation through the reactor core channel is 40.4 K. Table 1 and Figure 4 show the influence of core channel node number on the coolant temperature variation.

As can be seen from Table 1 and Figure 4, with the increase of node number in core channel, the fractional error of the calculated coolant temperature variation compared

with the design value first decreases and then increases, and it is about at the minimum level when the node number is 10. In addition, as the fractional error changes apparently with the node number, thus the node number of reactor core channel is a sensitive factor to the coolant temperature variation during the simulation of AP1000 NPP.

3.1.2. The Influence of Core Channel Node Number on the Reactor Pressure Drop. In reference to the design documents

TABLE 2: The influence of core channel node number on the reactor pressure drop.

Node number	Reactor inlet pressure (MPa)	Reactor outlet pressure (MPa)	Calculated reactor pressure drop (MPa)	Designed reactor pressure drop (MPa)	Fractional error
5	15.7545	15.3944	0.3601	0.43	16.26%
10	15.8342	15.4213	0.4129	0.43	3.98%
15	15.8947	15.3997	0.495	0.43	15.12%
20	15.9516	15.4006	0.551	0.43	28.14%
30	16.0739	15.6097	0.4642	0.43	7.95%
40	16.1328	15.6315	0.5013	0.43	16.58%
50	16.1675	15.6589	0.5086	0.43	18.28%

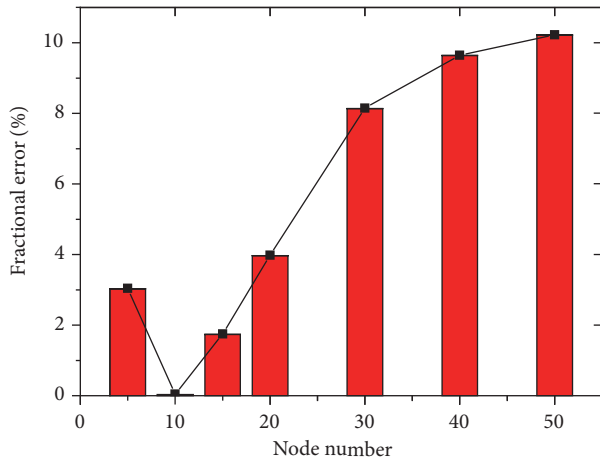


FIGURE 4: The influence of core channel node number on the fractional error between calculated and designed coolant temperature variation.

of AP1000 NPP (AP1000 Design Control Document, 2010), the overall drop of coolant pressure in the reactor is 0.43 ± 0.043 MPa. Table 2 and Figure 5 show the influence of core channel node number on the reactor pressure drop.

As can be seen from Table 2 and Figure 5, the fractional error between the calculated and the designed pressure drop in reactor varies irregularly with the increase of node number, and it is about at the minimum level when the node number is 10. Figure 5 also shows that the differences between fractional errors are evident, so the node number of reactor core channel is a sensitive factor to the pressure drop in reactor.

3.2. Sensitivity Evaluation on the Core Channel Number. According to Figure 3, the reactor core of the AP1000 NPP could be divided into different numbers of channels when simulating the system; thus sensitivity analysis could be implemented to value the importance of the core channel number. According to [12, 23, 24], the core channel number could be set at various fixed values from 1 to 20 listed in Tables 3 and 4.

3.2.1. The Influence of Core Channel Number on the Coolant Temperature Variation. In the best estimate analysis procedure of AP1000 Nuclear Power Plant, the core channel

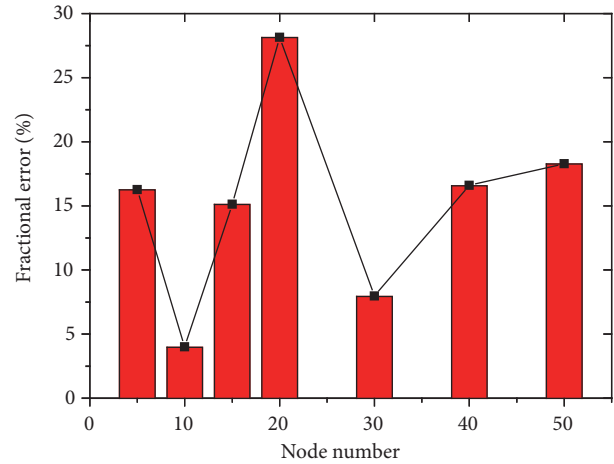


FIGURE 5: The influence of core channel node number on the fractional error between calculated and designed reactor pressure drop.

number has a direct influence on the complex level of simulation models. Overall, one-channel model is most simple, the three-channel model is most widely applied during simulation, and other multichannel models are more meticulous and comprehensive compared with these two models. Table 3 and Figure 6 show the influence of core channel number on the coolant temperature variation.

In reference to Table 3 and Figure 6, the fractional error between the calculated and the designed coolant temperature variation first decreases and then increases with the increase of channel number in the reactor core, and the fractional error is at the minimum level when the core channel number is 5. As the fractional error changes obviously with the core channel number, the reactor core channel number is sensitive to coolant temperature variation during the simulation process.

3.2.2. The Influence of Core Channel Number on the Reactor Pressure Drop. From Table 4 and Figure 7, the influence of core channel number on the reactor pressure drop could be evaluated.

Table 4 and Figure 7 see an abrupt change of the fractional error between calculated and designed pressure drop with

TABLE 3: The influence of core channel number on the coolant temperature variation.

Core channel number	Reactor inlet temperature (K)	Reactor outlet temperature (K)	Calculated temperature variation (K)	Designed temperature variation (K)	Fractional error
1	554.3814	594.6475	40.2661	40.4	0.33%
3	554.3778	594.6639	40.2861	40.4	0.28%
5	554.3637	594.7499	40.3862	40.4	0.03%
10	554.295	594.965	40.67	40.4	0.67%
20	554.2796	595.0128	40.7332	40.4	0.82%

TABLE 4: The influence of core channel number on the reactor pressure drop.

Core channel number	Reactor inlet pressure (MPa)	Reactor outlet pressure (MPa)	Calculated reactor pressure drop (MPa)	Designed reactor pressure drop (MPa)	Fractional error
1	15.8223	15.4123	0.4100	0.43	4.65%
3	15.8250	15.4166	0.4084	0.43	5.02%
5	15.8342	15.4213	0.4129	0.43	3.98%
10	15.891	15.6085	0.2825	0.43	34.30%
20	15.8647	15.5125	0.3522	0.43	18.09%

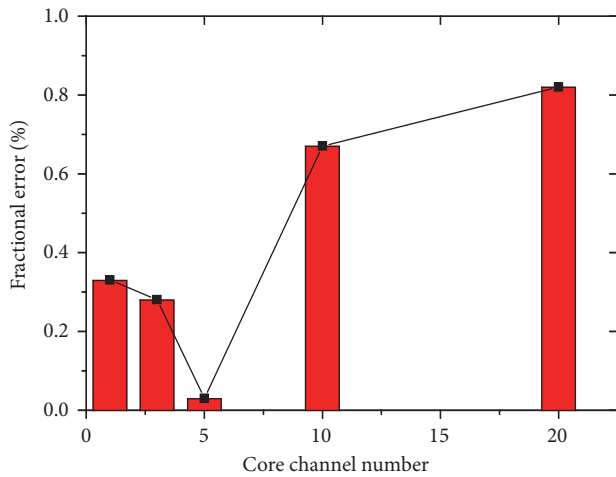


FIGURE 6: The influence of core channel number on the fractional error between calculated and designed coolant temperature variation.

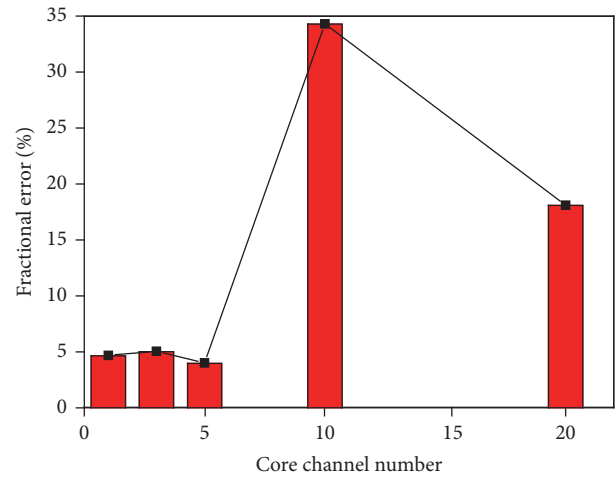


FIGURE 7: The influence of core channel number on the fractional error between calculated and designed reactor pressure drop.

the increase of core channel number in the reactor core. Consequently, core channel number is an influential factor to the reactor pressure drop. Based on the SPSS software, the statistical results of reactor inlet and outlet pressures are shown in Tables 5–8. As can be seen from the results, both of the reactor inlet pressure and reactor outlet pressure are normally distributed, the expectation values are 15.847 MPa and 15.474 MPa, and the standard deviations are 0.2 and 0.076, respectively.

3.3. Sensitivity Evaluation on the Pressurizer Node Number. During the AP1000 Nuclear Power Plant simulation process, pressurizer simulation also has a direct influence on the best estimation results of nuclear systems. In particular, within the primary system of Nuclear Power Plant, pressurizer is

TABLE 5: Statistical results of reactor inlet pressure.

N	
Valid	5
Missing	0
Mean	15.8474
Std. error of mean	.01324
Median	15.8342
Std. deviation	.02961
Variance	.001
Range	.07
Minimum	15.82
Maximum	15.89

employed to maintain the pressure of the primary loop; thus pressurizer node number may have a direct influence on the

TABLE 6: Nonparametric tests summary of reactor inlet pressure.

		Hypothesis test summary		
	Null hypothesis	Test	Sig.	Decision
1	The distribution of Reactor_inlet_pressure is normal with mean 15.847 and standard deviation 0.03	One-sample Kolmogorov-Smirnov test	.200 ^{1,2}	Retaining the null hypothesis

Asymptotic significance is displayed. The significance level is .05. 1: Lilliefors corrected. 2: this is a lower bound of the true significance.

TABLE 7: Statistical results of reactor outlet pressure.

N	
Valid	5
Missing	0
Mean	15.4742
Std. error of mean	.03837
Median	15.4213
Std. deviation	.08581
Variance	.007
Range	.20
Minimum	15.41
Maximum	15.61

code output. According to [12, 23, 24], the pressurizer node number could be set at various fixed values from 1 to 18 listed in Tables 9 and 10.

3.3.1. The Influence of Pressurizer Node Number on the Primary Loop Pressure. Table 9 and Figure 8 show the influence of pressurizer node number on the primary loop pressure.

As is shown in Table 9 and Figure 8, with the increase of node number in the pressurizer, the fractional error changes evidently and it is about at the minimum level when the node number is 6, so the pressurizer node number is a sensitive factor to the primary loop pressure.

3.3.2. The Influence of Pressurizer Node Number on the Coolant Temperature Variation. Table 10 and Figure 9 show the different coolant temperature variations under different pressurizer node numbers.

As is reported by Table 10 and Figure 9, the fractional error between the calculated and the designed temperature variation in the reactor core decreases significantly when the pressurizer node number is changed from 1 to 3 and then it varies wavelingly with the increase of node number in the pressurizer; thus the pressurizer node number is influential to the coolant temperature variation during the NPP simulation process. According to the statistical results of SPSS software shown in Tables 11 and 12, the reactor inlet temperature is normally distributed, and the expectation value and standard deviation are 552.534 and 0.2.

3.3.3. The Influence of Pressurizer Node Number on the Reactor Pressure Drop. As is demonstrated above, the pressurizer has a direct influence on the primary loop pressure, so the

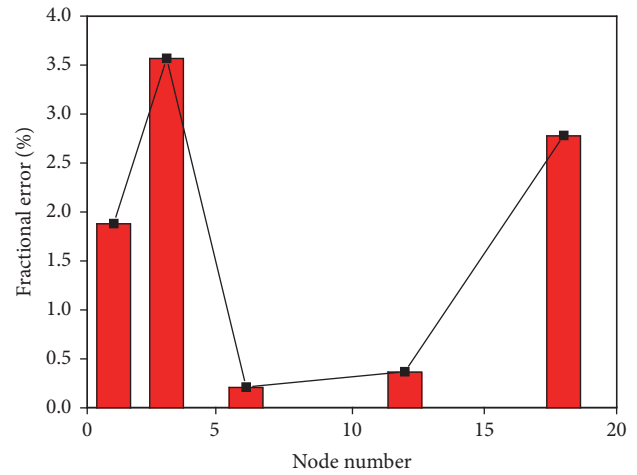


FIGURE 8: The influence of pressurizer node number on the fractional error between calculated and designed primary loop pressure.

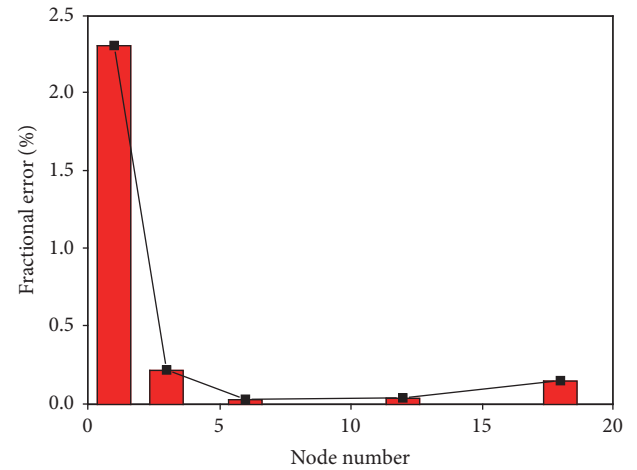


FIGURE 9: The influence of pressurizer node number on the fractional error between calculated and designed coolant temperature variation.

pressurizer node number may also have some effects on the reactor pressure drop, which are shown in Table 13 and Figure 10.

According to Table 13 and Figure 10, with the increase of the fractional error between the calculated and the designed reactor pressure drop varies distinctly especially when the pressurizer node number is altered from 12 to 18, so the pressurizer node number is a sensitive factor to the reactor

TABLE 8: Nonparametric tests summary of reactor outlet pressure.

Null hypothesis		Hypothesis test summary	Sig.	Decision
		Test		
1	The distribution of Reactor_outlet_pressure is normal with mean 15.474 and standard deviation 0.09	One-sample Kolmogorov-Smirnov test	.076 ¹	Retaining the null hypothesis

Asymptotic significance is displayed. The significance level is .05. 1: Lilliefors corrected.

TABLE 9: The influence of pressurizer node number on the primary loop pressure.

Node number	Calculated pressure of primary loop (MPa)	Designed pressure of primary loop (MPa)	Fractional error
1	15.7916	15.5	1.88%
3	14.9463	15.5	3.57%
6	15.5322	15.5	0.208%
12	15.5565	15.5	0.365%
18	15.0693	15.5	2.78%

TABLE 10: The influence of pressurizer node number on the coolant temperature variation.

Node number	Reactor inlet temperature (K)	Reactor outlet temperature (K)	Calculated temperature variation (K)	Designed temperature variation (K)	Fractional error
1	550.56	591.8925	41.3325	40.4	2.31%
3	551.7598	592.07125	40.3115	40.4	0.22%
6	554.3637	594.7499	40.3862	40.4	0.03%
12	554.3674	594.7526	40.3852	40.4	0.04%
18	551.62	591.96	40.34	40.4	0.15%

TABLE 11: The statistical results of reactor inlet temperature.

N	
Valid	5
Missing	0
Mean	552.5342
Std. error of mean	.77591
Median	551.7598
Std. deviation	1.73498
Variance	3.010
Range	3.81
Minimum	550.56
Maximum	554.37

pressure drop in the simulation process. In addition, the fractional error is about at the minimum level when the pressurizer node number is 3.

3.4. Sensitivity Evaluation on Feedwater Temperature. As is known, the feedwater in the secondary side is heated by the primary loop coolant; thus the steam is produced and used to propel the turbines after flowing through the moisture separator-dryer. Consequently, the feedwater temperature has direct influences on the steam properties.

3.4.1. The Influence of Feedwater Temperature on the Steam Output. According to [26], the distribution of feedwater

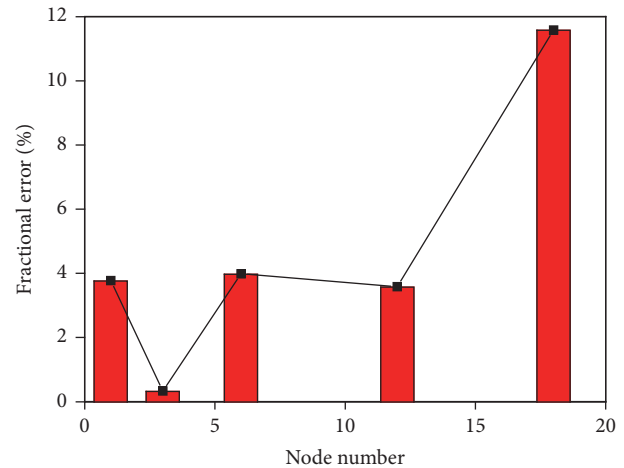


FIGURE 10: The influence of pressurizer node number on the fractional error between calculated and designed reactor pressure drop.

temperature in AP1000 NPP is approximately uniform distribution as $U(494.85, 504.85)$. In addition, the feedwater with higher initial temperature is more likely to generate steam; thus the feedwater temperature may have some effects on the steam properties. Table 14 and Figure 11 show the influence of feedwater temperature on the steam output.

As can be seen from Table 14 and Figure 11, the fractional error between the calculated and the designed steam output

TABLE 12: Nonparametric tests summary of reactor inlet temperature.

Hypothesis test summary				
	Null hypothesis	Test	Sig.	Decision
1	The distribution of Reactor_inlet_temperature is normal with mean 552.534 and standard deviation 1.73	One-sample Kolmogorov-Smirnov test	.200 ^{1,2}	Retaining the null hypothesis

Asymptotic significance is displayed. The significance level is .05. 1: Lilliefors corrected. 2: this is a lower bound of the true significance.

TABLE 13: The influence of pressurizer node number on the reactor pressure drop.

Node number	Reactor inlet pressure (MPa)	Reactor outlet pressure (MPa)	Calculated reactor pressure drop (MPa)	Designed reactor pressure drop (MPa)	Fractional error
1	15.8587	15.4125	0.4462	0.43	3.77%
3	15.4805	15.0491	0.4314	0.43	0.33%
6	15.8342	15.4213	0.4129	0.43	3.98%
12	15.8819	15.4365	0.4454	0.43	3.58%
18	15.4929	15.0131	0.4798	0.43	11.58%

TABLE 14: The influence of feedwater temperature on the steam output.

Feedwater temperature (K)	Calculated steam output (kg/s)	Designed steam output (kg/s)	Fractional error
494.85	970.2	943.6111	2.818%
496.85	970.19	943.6111	2.817%
499.85	970.18	943.6111	2.816%
502.85	970.18	943.6111	2.816%
504.85	970.17	943.6111	2.815%

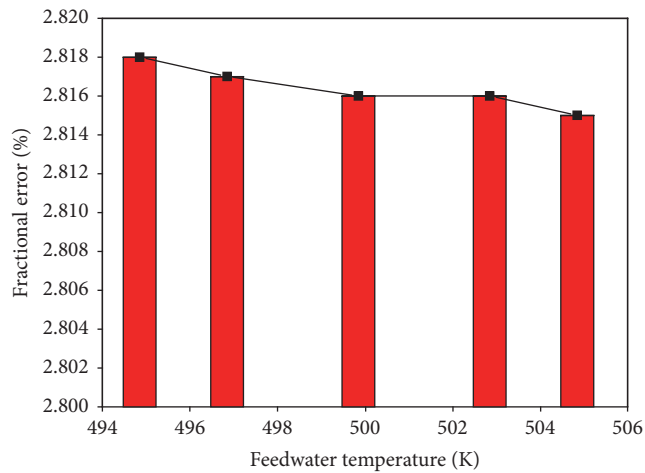


FIGURE 11: The influence of feedwater temperature on the fractional error between calculated and designed steam output.

stays steady with the increase of feedwater temperature; thus the feedwater temperature is not a sensitive factor to the steam output.

3.4.2. The Influence of Feedwater Temperature on the SG Outlet Temperature. Table 15 and Figure 12 show the variation of SG outlet temperature under different feedwater temperatures.

In reference to Table 15 and Figure 12, the fractional error between the calculated and the designed SG outlet temperature rises straightforward with the increase of feedwater temperature, so the feedwater temperature is sensitive to the SG outlet temperature.

3.4.3. The Influence of Feedwater Temperature on the SG Outlet Pressure. Table 16 and Figure 13 show the variations of SG outlet pressures and related fractional errors compared with the design value under different feedwater temperatures.

According to Table 16 and Figure 13, the fractional error between the calculated and the designed SG outlet pressure rises straightforward with the increase of feedwater temperature; thus the feedwater temperature is a sensitive factor to the SG outlet pressure.

3.5. Sensitivity Evaluation on the Cross-Flow Model. Overall, the coolant in the reactor core flows in the axial direction, while there exists cross-flow between the different channels. In the simulation process of the primary system based on RELAP5 platform, the single-channel model treats the core as an integration system; the water flows homogeneously in the core channel; in this case, the cross-flow phenomenon could not be demonstrated. In regard to the multichannel core model, there exists cross-flow phenomenon in reality, and whether considering the cross-flow between different channels or not may have different influences on the coolant

TABLE 15: The influence of feedwater temperature on the SG outlet temperature.

Feedwater temperature (K)	Calculated SG outlet temperature (K)	Designed SG outlet temperature (K)	Fractional error
494.85	546.12	546.1	0.00367%
496.85	546.14	546.1	0.00732%
499.85	546.18	546.1	0.0146%
502.85	546.22	546.1	0.0220%
504.85	546.24	546.1	0.0256%

TABLE 16: The influence of feedwater temperature on the SG outlet pressure.

Feedwater temperature (K)	Calculated SG outlet pressure (MPa)	Designed SG outlet pressure (MPa)	Fractional error
494.85	5.766085	5.764	0.0362%
496.85	5.76833	5.764	0.0751%
499.85	5.771655	5.764	0.133%
502.85	5.774995	5.764	0.191%
504.85	5.777175	5.764	0.229%

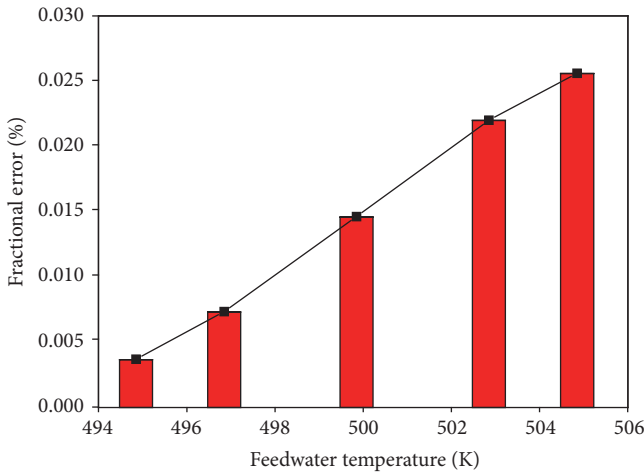


FIGURE 12: The influence of feedwater temperature on the fractional error between calculated and designed SG outlet temperature.

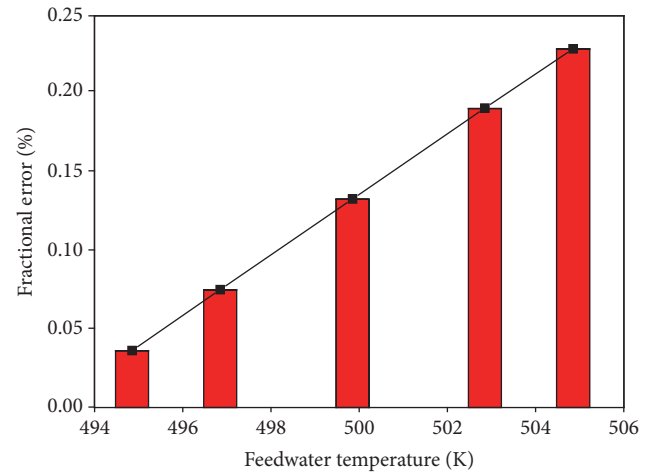


FIGURE 13: The influence of feedwater temperature on the fractional error between calculated and designed SG outlet pressure.

flow, coolant temperature, pressure drop in the reactor core, and some other system parameters.

3.5.1. The Influence of Cross-Flow Model on the Coolant Temperature Variation. According to the design parameters of AP1000 Nuclear Power Plant, the water temperature rise through the reactor coolant channels is 42.83 K. Table 17 shows the coolant temperature changes whether applying cross-flow model or not on the same conditions of boundary parameters, flow resistance coefficients, surface roughness factors, and so on.

As can be seen from Table 17, whether taking the cross-flow model into consideration in the simulation process of primary system or not, the fractional error nearly stays the same; thus the cross-flow model is not a sensitive factor to the coolant temperature variation in the reactor core.

3.5.2. The Influence of Cross-Flow Model on the Reactor Core Pressure Drop. In reference to the design documents of

AP1000 Nuclear Power Plant, the designed pressure drop through the reactor core is 0.27 ± 0.028 MPa, so the upper limit could be set as 0.298 MPa. Table 18 shows the pressure drop in the reactor core whether applying the cross-flow model or not on the same conditions of boundary parameters, flow resistance coefficients, surface roughness factors, and so on.

As can be seen from Table 18, whether taking the cross-flow model into consideration in the simulation process of primary system or not, the difference between two fractional errors is insignificant; thus the cross-flow model is not a sensitive factor to the pressure drop in the reactor core.

3.5.3. The Influence of Cross-Flow Model on the Core Coolant Mass Flow. In reference to the design documents of AP1000 Nuclear Power Plant, the core coolant mass flow is 14301.0 kg/s. Table 19 shows the core coolant mass flow whether applying cross-flow model or not on the same conditions of boundary parameters, flow resistance coefficients, surface roughness factors, and so on.

TABLE 17: The influence of cross-flow model on the coolant temperature variation.

Cross-flow model	Core inlet temperature (K)	Core outlet temperature (K)	Calculated temperature variation (K)	Designed temperature variation (K)	Fractional error
None	554.4102	596.6464	42.2362	42.83	1.39%
Having	554.4102	596.6468	42.2366	42.83	1.39%

TABLE 18: The influence of cross-flow model on the pressure drop in the reactor core.

Cross-flow model	Core inlet pressure (MPa)	Core outlet pressure (MPa)	Calculated pressure drop in reactor core (MPa)	Designed pressure drop in reactor core (MPa)	Fractional error
None	15.9615	15.6468	0.3147	0.298	5.60%
Having	15.9618	15.6470	0.3148	0.298	5.63%

TABLE 19: The influence of cross-flow model on the core coolant mass flow.

Cross-flow model	Calculated core coolant mass flow (kg/s)	Designed core coolant mass flow (kg/s)	Fractional error
None	14304.73	14301.0	0.026%
Having	14304.46	14301.0	0.024%

As can be seen from Table 19, whether taking the cross-flow model into consideration in the simulation process of primary system or not, the difference between two fractional errors is insignificant; thus the cross-flow model is not a sensitive factor to the coolant mass flow in the reactor core.

3.6. Sensitivity Evaluation on the Cross-Flow Model during Loss of Power Transient. The loss of power transient is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. During the loss of power transient, core decay heat removal is normally accomplished by the startup feedwater system if available, which is started automatically when low levels occur in the Steam Generator. If the startup feedwater system is not available, emergency core decay heat removal is provided by the PRHR heat exchanger. Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant and PRHR loops [27, 28]. In this case, the cross-flow model may have effect on the mass flow of coolant through the core channel, Peak Cladding Temperature (PCT), and some other parameters.

3.6.1. The Influence of Cross-Flow Model on Core Coolant Mass Flow. Figure 14 shows the coolant mass flow in the reactor core whether applying cross-flow model or not during the loss of power transient, on the same conditions of boundary parameters, flow resistance coefficients, surface roughness factors, and so on.

As can be seen from Figure 14, whether taking the cross-flow model into consideration in the simulation process of primary system or not, the difference between the coolant mass flows on two conditions is insignificant; thus the cross-flow model is not a sensitive factor to the coolant mass flow in the reactor core.

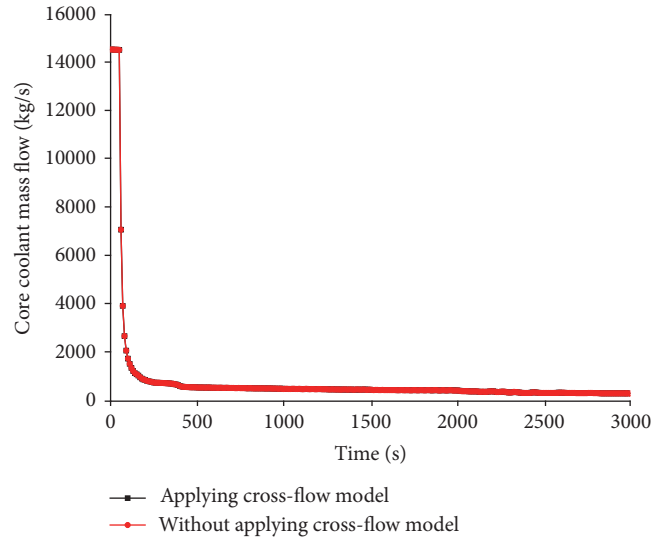


FIGURE 14: The influence of cross-flow model on core coolant mass flow.

3.6.2. The Influence of Cross-Flow Model on PCT. Figure 15 shows the PCT of applying cross-flow model or not during the loss of power transient, on the same conditions of boundary parameters, flow resistance coefficients, surface roughness factors, and so on.

According to Figure 15, whether taking the cross-flow model into consideration in the simulation process of primary system or not, there is a slight difference between the PCT on two conditions; thus the cross-flow model is not a sensitive factor to the Peak Cladding Temperature.

4. Conclusions

The best estimation process of AP1000 NPP requires proper selections of parameters and models so as to obtain the

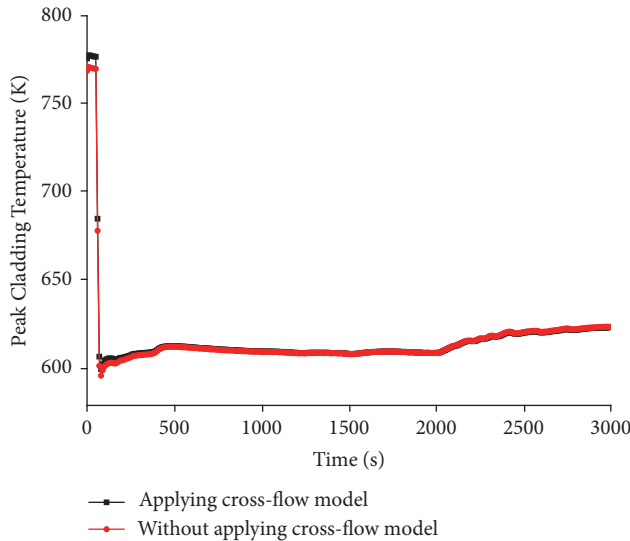


FIGURE 15: The influence of cross-flow model on PCT.

most accurate results compared with the actual design parameters. Based on the thermal-hydraulic system code RELAP5/MOD3.2, sensitivity analyses on various of the parameters and models have been performed in this paper to provide reference to the simulation process of nuclear power systems. Main results obtained in this study are summarized as follows.

- (1) During the best estimation process of AP1000 NPP, the simulation results, especially the coolant temperature variation and pressure drop in reactor, are sensitive to both the number of core channels and the node number of each channel within the best estimation model. With the variation of core channel number, the reactor inlet and outlet pressures and the reactor inlet temperature are normally distributed. In addition, the best estimation results of coolant temperature variation and pressure drop through the AP1000 NPP reactor are about the most accurate when the node number and channel number are 10 and 5, respectively, compared with the other conditions.
- (2) The pressurizer node number has apparent influences on the primary loop pressure, coolant temperature variation, and pressure drop in reactor. In addition, the best estimation results of primary loop pressure and coolant temperature are about the most accurate when the node number is 6, while the corresponding node number of reactor pressure drop is 3.
- (3) Both the SG outlet temperature and pressure are susceptible to feedwater temperature, but the steam output is on the contrary. Besides, both of the fractional errors of SG outlet temperature and pressure compared with the corresponding designed values rise straightforward with the increase of feedwater temperature within its range.

- (4) Whether considering the cross-flow model or not, the simulation results of AP1000 NPP, such as coolant temperature variation, pressure drop, and coolant mass flow, remain stable both at steady state and during loss of power transient.

Nomenclature

X_i : i th input variable
 x_j : j th input variable
 x_{ij} : i th sample of the j th input variable
 y_i : i th model output
 N : Number of samples
 Y : Response
 y : Model output
 \bar{x}_j : Mean value of the j th input variable
 \bar{y} : Mean value of model output.

Abbreviation

CC: Correlation coefficient
 GSA: Global sensitivity analysis
 NPP: Nuclear Power Plant
 PCT: Peak Cladding Temperature
 PRHR: Passive Residual Heat Remove
 RCC: Rank correlation coefficient
 RELAP: Reactor Excursion and Leak Analysis Program
 SA: Sensitivity analysis
 SG: Steam Generator.

Conflicts of Interest

The authors declare that they have no conflicts of interest.

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