Assessment of Coupled Effect of Steam Pipes on Seismic Analysis of Reactor Coolant System

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Abstract. Reactor coolant system is connected with auxiliary pipes, surge pipes and steam pipes. In order to facilitate the dynamic analysis of reactor coolant system, it is necessary to decouple the pipes connected with the reactor coolant system. Although some international standards stipulating requirements for the decoupling of subsystems from the main system, in order to further explore the influence of decoupling steam pipes, the seismic response of reactor coolant system with or without steam pipes is simulated by using ANSYS program. Firstly, the modal analysis of reactor coolant system is carried out to analysis the influence of steam pipes on the resonant frequency of reactor system. Then, the standardized seismic dynamic analysis of reactor system with or without steam pipes is carried out. The nozzle load (inlet and outlet nozzle load of main device, such as reactor pressure vessel, steam generator, pump) and support load (of main device such as reactor pressure vessel, steam generator, pump) are output and compared. According to nozzle load and support load, it can be seen that the steam pipes have little influence on the seismic analysis of reactor coolant system, i.e. steam pipes can be decoupled in the seismic analysis of reactor coolant system.

Keywords. Reactor coolant system, dynamic analysis, seismic response, decouple, steam pipes.

1. Introduction

Generally, the structural mass, stiffness, and damping characteristics of the subsystem differ greatly from the main structure, so the primary and secondary combined systems have much more complex dynamic characteristics than the general structure. There is interaction between the primary and secondary structures, a certain correlation between the excitations of each support point, and a certain correlation between the vibration modes as well. In the dynamic analysis of primary and secondary systems, more complex problems may be encountered, such as frequency tuning and interaction between primary and secondary systems, non-classical damping [1] of primary and secondary combined systems, decoupling criteria, and the problem of mass loss in seismic response of quadratic structure solved by modal superposition method.

In the design analysis of nuclear power plants, it is unrealistic to include all equipment in coupled model. The reactor coolant system is connected to auxiliary pipelines [2], surge pipes, steam pipes and other pipelines. To facilitate the dynamic analysis of the reactor coolant system, it is necessary to decouple the connected pipelines with reference to certain decoupling criteria. The decoupling criteria in the current standard of nuclear power plant seismic design cannot meet the actual construction needs. Therefore, in this study the steam pipes were selected to decouple, the seismic response of the coupled and decoupled reactor system were compared to provide data support for simplifying the
seismic analysis of the reactor system [1-6]. In this study, the ANSYS program was used to perform the standardized seismic analysis of the reactor coolant system with and without steam pipes. The seismic response of the reactor coolant system at interested locations is extracted: the main equipment inlet and outlet nozzle load and the main equipment supporting load. By comparing the results of the seismic analysis mentioned above, the influence of steam pipes on reactor system seismic response is given, and the rationality of steam pipes decoupling is verified. These works in the paper is of positive significance for the revision of the decoupling criterion in the seismic design code.

2. Calculation Model

Based on a nuclear power plant, the study of the influence of steam pipes on the seismic response of the reactor coolant system was carried out. The schematic diagram of the reactor coolant system is shown in figure 1 below. In the calculation model, the influence of reactor building is considered by using multi-point spectrum analysis method. The seismic acceleration spectrum is applied to each location of the reactor system which is connected to the building.

![Figure 1. Schematic diagram of the reactor coolant system.](image)

3. Decoupling Argument

3.1. Decoupling Criteria

The Standard Review Plan (SRP) issued by the US Nuclear Regulatory Commission (NRC) provides guidelines for acceptance or approval by evaluators, including Section 3.7.2 for seismic design analysis of nuclear power plants. The criteria for subsystem decoupling from the primary system are defined in US NRC Section 3.7.2. USNRC 3.7.2 [7-8] specifies two indicators of mass and frequency for the decoupling of subsystems from the main system, namely:

\[
R_m = \frac{\text{Subsystem quality}}{\text{Total mass of support system}}, \quad R_f = \frac{\text{Subsystem fundamental frequency}}{\text{Support system dominant frequency}}
\]  

(1)

The subsystem can be decoupled from the main system in seismic analysis while the following conditions are met:

(1) When \( R_m < 0.01 \), the subsystem can be decoupled regardless of the value of \( R_f \);
(2) When \( 0.01 \leq R_m < 0.1 \), and \( R_f \geq 1.25 \) or \( R_f \leq 0.8 \), the subsystem can be decoupled.
The above decoupling criterion [9-13] can be understood that the subsystem can be decoupled when the following conditions are met: (1) The subsystem mass is less than 1/100 of the main system; (2) the main system mass is between 10 and 100 times the subsystem mass, and the frequency of the primary system is offset from the frequency of the subsystem by approximately 20% (i.e., the subsystem frequency is detuned from the primary system).

3.2. Steam Pipes Decoupling Argument
This study applies the modal strain energy theory and the effective mass of the dynamic system modal analysis, and demonstrates whether the steam pipes can be decoupled from the reactor coolant system during seismic analysis according to the decoupling criteria of SRP 3.7.2. The calculation result is that the $R_{in}$ value of the steam pipes is 0.0348, which does not satisfy the first clause of the decoupling criterion. Thus, the second clause of the decoupling criterion should be considered in combination with the modal information.

3.3. Dominant Frequency of Main System
The dominant frequency of the primary system is determined by the effective mass in each direction. The importance of each mode in each direction can be determined by the effective mass. The modes of the main structure and the sub-structure are distinguished by the modal strain energy theory and the effective mass of the dynamic system modal analysis. It can be indicated that the energy contribution of rotational freedom in the case of resonance is very small by analyzing the modal strain energy distribution of different degrees of freedom of each component. Hence, when considering the dominant frequency, only the effective mass distribution of each translational direction is considered.

The dominant mode of the main system is extracted in the first 300 modes of the system. The mode of which the effective mass in each direction accounts for more than 5% of the total mass of the main system and the proportion of the participating mass in each direction is extracted. The dominant frequency in the $X$, $Y$, and $Z$ directions of the main system coupled to the steam pipes can be obtained by the effective mass in each direction.

The distribution of the weighted mode strain energy of different components of the main system is extracted, and the modal strain energy weight is weighted according to the $X$, $Y$ and $Z$ three-direction effective mass. It can be seen that the modal strain energy of the reactor coolant system accounts for the majority of the energy in the dominant mode. Therefore, when examining the local mode of the steam pipes, the modal strain energy should be considered to obtain the fundamental frequency of the steam pipes.

3.3.1. Main Mode and Fundamental Frequency of Steam Pipes. In this study, the modal superposition method in the multi-point spectrum analysis function of the finite element analysis software ANSYS was used for seismic response spectrum analysis. Before the spectrum analysis, the modal analysis was first carried out. The modal analysis of the BLOCK LANCZOS method was used to extract the first 300 modes, so that the effective modal masses in the $X$, $Y$ and $Z$ directions was extracted. And the effective modal masses in three directions all exceed or reach 99% of the total mass of the model.

3.3.2. Main Steam Pipes Decoupling Judgment. The numerator and denominator of the $R_t$ value of the steam pipes are given above. Table 1 shows the $R_t$ value distribution of the steam pipes when considering the dominant frequency of the main system in different directions. The maximum value is 0.5860, which satisfies the decoupling condition of $R_t \leq 0.8$.

| Table 1. Ratio of the steam pipes fundamental frequency to the main system dominant frequency. |
|-----------------|-----------|-----------|-----------|
| Frequency / Hz  | $X$ direction | $Y$ direction | $Z$ direction |
| $R_t$           | 0.4756    | **0.5860** | 0.1694    |
3.4. Data Comparison before and after Decoupling

In order to further verify the correctness of the decoupling criterion, this study extracted the natural frequencies of the reactor coolant system before and after decoupling of the steam pipes, the main equipment inlet and outlet pipe load and the main equipment support load for comparative analysis.

3.4.1. Natural Frequencies of Reactor Coolant System. Table 2 shows the natural frequencies of the main equipments in the reactor coolant system before and after steam pipes decoupling. According to the data comparison, the natural frequency of the main equipments (except the steam generator directly connected to the steam pipes) after decoupling of the steam pipes changes less than 0.06%, indicating that the steam pipes decoupling has little effect on the natural frequency of the reactor coolant system.

Table 2. Natural frequency changes of the main device before and after decoupling.

| Order | Couple /Hz | Decouple /Hz | Main device | Changes of Natural Frequency |
|-------|------------|--------------|-------------|-----------------------------|
| 1     | 6.508      | 6.350        | Steam Generator | -2.42%                      |
| 2     | 6.526      | 6.350        | Steam Generator | -2.69%                      |
| 3     | 6.562      | 6.387        | Steam Generator | -2.67%                      |
| 4     | 7.071      | 7.071        | (Surge Pipe)   | 0.00%                       |
| 5     | 7.221      | 7.221        | Pump          | 0.00%                       |
| 6     | 7.221      | 7.221        | Pump          | 0.00%                       |
| 7     | 7.222      | 7.221        | Pump          | -0.01%                      |
| 8     | 8.036      | 8.218        | Steam Generator | 2.27%                      |
| 9     | 8.036      | 8.218        | Steam Generator | 2.26%                      |
| 10    | 8.229      | 8.219        | Steam Generator | -0.12%                      |
| 11    | 8.434      |              |              |                             |
| 12    | 8.962      | 8.962        | Pump          | 0.00%                       |
| 13    | 8.962      | 8.962        | Pump          | 0.00%                       |
| 14    | 8.968      | 8.968        | Pump          | 0.00%                       |
| 15    | 9.154      | 9.154        | Pressurizer   | 0.00%                       |
| 16    | 9.284      | 9.284        | Pressurizer   | 0.00%                       |
| ...... | ......      | ......        | .....         | .....                        |
| 29    | 15.342     | 15.344       | RPV           | 0.01%                       |
| 30    | 15.348     | 15.349       | RPV           | 0.01%                       |
| ...... | ......      | ......        | .....         | .....                        |

3.4.2. Main Equipment Inlet and Outlet Nozzle Load. In order to further explore the influence of steam pipes decoupling, in this study the load of each main equipment nozzle (reactor pressure vessel inlet and outlet nozzle, steam generator inlet and outlet nozzle, pump inlet and outlet nozzle) was extracted. Under the action of the reactor structure response spectrum of the SL-2 condition [1], the maximum absolute values of the inlet and outlet nozzles are shown in table 3. According to the data comparison, after the steam pipes is decoupled, the change of the nozzle load of the main equipment is mostly below 5%. It is worth mentioning that although the pressure vessel outlet nozzle and the pump inlet and outlet nozzles have a large percentage change in load components, the absolute load value is small, and the load difference is less than 40 (10^4 N). This small load fluctuation often occurs under normal operating conditions of the reactor coolant system and it has no effect on the safe operation of the reactor. Therefore, it can be concluded that the steam pipes decoupling has little effect on the inlet and outlet nozzle load of the main coolant of the reactor coolant system.
Table 3. Load changes of the inlet and outlet of the main equipment before and after decoupling (unit: force $10^4$ N, torque $10^4$ N·m).

| Load    | Outlet Decouple | Outlet Couple | Load Change (%) | Inlet Decouple | Inlet Couple | Load Change (%) |
|---------|-----------------|---------------|-----------------|----------------|--------------|-----------------|
| Reactor pressure vessel | FX | 538.8 | 514.9 | 4.6% | 116.2 | 78.0 | 49.0% |
|         | FT | 627.5 | 602.0 | 4.2% | 116.9 | 90.7 | 28.9% |
|         | MX | 335.7 | 331.7 | 1.2% | 113.9 | 102.1 | 11.5% |
|         | MT | 305.4 | 302.2 | 1.1% | 206.4 | 196.4 | 5.1% |
| Pump    | FX | 118.3 | 86.4 | 36.9% | 70.9 | 57.8 | 22.8% |
|         | FT | 114.1 | 89.4 | 27.6% | 138.7 | 123.3 | 12.5% |
|         | MX | 144.0 | 143.2 | 0.5% | 311.2 | 306.7 | 1.5% |
|         | MT | 217.1 | 216.3 | 0.4% | 317.4 | 309.9 | 2.4% |
| Steam generator | FX | 115.1 | 85.8 | 31.4% | 503.3 | 503.3 | 0.0% |
|         | FT | 150.0 | 146.9 | 2.1% | 605.5 | 579.8 | 4.4% |
|         | MX | 234.7 | 233.2 | 0.6% | 478.3 | 452.8 | 5.6% |
|         | MT | 261.3 | 252.8 | 3.4% | 429.8 | 407.0 | 5.6% |

3.4.3. Main Equipment Support Load. The reactor coolant system supports include reactor pressure vessel supports, steam generator supports, pump supports. Compare the maximum absolute values of the individual support of each device, as shown in table 4. According to the data comparison, after the steam pipes is decoupled, the load change of the main equipment support position is basically less than 6%. Although the load variation at the vertical support position of the main pump is large, the load variation at the horizontal support of the pump and at all the support positions of the steam generator and the reactor pressure vessel is small. Therefore, it can be considered that the decoupling of the steam pipes has little influence on the load of support position of the main equipment of the reactor coolant system.

Table 4. Support load changes of main device before and after decoupling (unit: force $10^4$ N).

| Main Device | Main Device Number | Support Location | Decouple | Couple | Load Change (%) |
|-------------|--------------------|------------------|----------|--------|-----------------|
| Reactor pressure vessel | 1 | Vertical | 858.7 | 858.7 | 0.0% |
|             |                   | Lower part, horizontal | 384.2 | 364.1 | 5.5% |
|             |                   | Upper part, horizontal | 529.3 | 502.6 | 5.3% |
|             |                   | Vertical | 860.5 | 861.5 | -0.1% |
| Steam generator | 2 | Lower part, horizontal | 384.4 | 369.4 | 4.0% |
|             |                   | Upper part, horizontal | 529.6 | 531.7 | -0.4% |
|             |                   | Vertical | 860.7 | 861.4 | -0.1% |
|             | 3 | Lower part, horizontal | 384.2 | 369.0 | 4.1% |
|             |                   | Upper part, horizontal | 529.8 | 531.5 | -0.3% |
|             |                   | Vertical | 414.6 | 359.3 | 15.4% |
| Pump | 1 | Lower part, horizontal | 349.7 | 351.2 | -0.4% |
|             | Vertical | 414.2 | 356.3 | 16.3% |
|             | 2 | Lower part, horizontal | 349.7 | 347.1 | 0.7% |
|             | Vertical | 414.9 | 357.0 | 16.2% |
|             | 3 | Lower part, horizontal | 349.6 | 347.9 | 0.5% |
| Reactor pressure vessel | -- | Vertical | 187.6 | 184.6 | 1.6% |
|             | Horizontal | 730.1 | 705.6 | 3.5% |
4. Conclusions

In general, the interaction of the primary system and subsystem reduces the response of the substructure. Decoupling the steam pipes from the reactor coolant system, ignoring its interaction with the primary system is a conservative measure. But sometimes the result of such calculations may result in a decoupled subsystem response of more than a few hundred percent of the actual response. In this study, when using the modal superposition method, in order to reduce the amount of calculation, and to avoid the numerical problems which may be encountered in the calculation of higher-order modes, the contribution of the 100 lower-order modes to the response is selected to make the effective modal mass in all directions reach 99% of the total mass of the model. This is usually reasonable because the effects of higher-order modes on the response are usually small. However, the modal displacement method used above completely ignores the influence of the higher-order modes on the response, which may lead to errors in the calculation response, namely, “loss of mass”. And thereby the subsystem response accuracy are greatly reducing. The treatment of “loss of mass” can be further studied. Considering that this study mainly discusses the effect of decoupling on the response of the main device of reactor coolant system, this paper does not discuss “loss of mass”.

In this study the standardized seismic response of the reactor coolant system before and after decoupling steam pipes from reactor coolant system were analyzed, and the natural frequency of the main equipment before and after decoupling, the inlet and outlet connection load, and the supporting load were extracted. The conclusion is obtained that the steam pipes have little influence on the seismic analysis results of the reactor coolant system, and the feasibility of decoupling the steam pipes from the reactor coolant system is clarified. The decoupling criterion in this paper can be extended to other systems, such as pressurizer and surge pipes can be decoupled from the reactor coolant system.

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