Research on fire PRA for NPPs

SUN Feng[1,2], ZHAO Bo[1,2], and ZHAO Qingnan[1]

1. *Fundamental Science on Nuclear Safety and Simulation Technology Laboratory, Harbin Engineering University (smlink260@126.com)*
2. *China Nuclear Power Engineering, Beijing 100840, China*

Abstract: Fire is one of the threats to nuclear power plants safety. Fire analysis, using probabilistic risk assessment (PRA), can find out the weakness of the plant and improve its design. Based on the study on fire PRA methodology widely used across the world, a fire PRA is developed for a typical second-generation pressurized water plant and the results show that the core damage frequency induced by fire is 4.03×10-6/(reactor year). After that, sensitivity analysis is performed and the influence of human error and quantitative screening value are discussed.

Keyword: fire; probabilistic risk assessment; core damage frequency

1 Background

Fukushima accident was a wake-up call to the global nuclear safety, which led some countries like the United States, France, Germany and other countries pay more attention to serious nuclear power plant accident, emergency and hazard events analysis. National Nuclear Safety Administration (the nuclear safety authority in China) immediately initiated a series of safety inspection work and put forward safety improvement requirements for NPPs under construction. Hazard events PRA including internal fire are treated as one of the research work for a long time. Quantitative risk should be given for new built NPPs according to the new published guidance document “safety requirements for new generation nuclear power plant and prospective targets of 2020”. As the results, the practice of full scope PRA has become indispensable.

Fire is one of the NPPs safety risks. Fire protection is a key problem and must be considered for NPPs. At present, nuclear power plant fire safety assessment method mainly based on deterministic and probabilistic technologies. Deterministic fire safety assessment, namely, Fire Hazard Analyses (FHA) is based on the concept of “defense in depth” of NPPs fire protection. However, Fire Probabilistic Risk Assessment (FPRA) is based on deterministic and reliability evaluation technologies, which can give quantitative risk of fire through systematic analysis, also proposes further improvement and optimization suggestion for fire protection design and management.[1]

FPRA is a comprehensive project involving a number of specialty fields, which includes NPPs fire protection design, cable routing, circuit failure analysis and fire scenario analysis. FPRA has some characteristics of heavy workload, long development cycle, more technical difficulties and closely with design. In order to obtain risk contribution of Fire PRA for specific NPP, This paper selects PWR NPP as the object of study, and screens out ignition source, develops fire risk model and simulates fire scenario by using guidelines and documents from IAEA, NRC and ERPI.

2 Development status in China and abroad

In 1975, with the US WASH-1400 coming out, study on fire PRA was started. The world carried out a great amount of work on PRA for NPPs. Fire PRA has been regarded as an important part of PRA for NPPs.

In the late 80s, NRC required all NPPs to develop Individual Plant Examination of External Events (IPEEE). Most NPPs adopts PRA method to do IPEEE. EPRI issued semi-quantitative method --FIVE in April 1992(The fire-induced vulnerability evaluation, FIVE, methodology，EPRI TR-100370), [2] which belongs to deterministic and probabilistic combined methodology. In 1995, EPRI issued Fire PRA Implementation Guide， EPRITR-105928,[3]

Received date: February 16, 2017
(Revised date: June 8, 2017)

which meets the goals of IPEEE, finally accepted by NRC.

In 1998, IAEA issued the report Treatment of internal fires in PRA for nuclear power plants(Safety Report Series No.10), [4] which aims to guide how to develop FPRA for NPPs.

In 2005, NRC and EPRI officially published NUREG/CR-6850 (Fire PRA Methodology for Nuclear Power Facilities),[5]which is the most comprehensive method for Fire PRA now.[6] Most countries refer NUREG/CR-6850 to conduct Fire PRA.

In China, Fire PRA is in a development stage. Many new NPPs are actively carrying out Fire PRA analysis at present. Daya Bay plant, Fuqing plant and Hainan plant are developing fire PRA referring to NUREG/CR-6850. So the study of Fire PRA and its applicability in Chinese NPPs have an important significance.

3 The analysis process of fire PRA

Fire PRA in this text mainly refers to ASME-RA-Sa-2009[7] and NUREG/CR-6850. NUREG/CR-6850 is a general method of PRA. During the course of developing Fire PRA, we need make sure Fire PRA analysis tasks and process suitable for NPP based on the characteristics of the NPP(*e.g.* fire protection design, fire management, *etc.*) and possible PRA application needs. The process of the fire PRA is shown in Fig.1. The process mainly contains three stages including qualitative analysis, quantitative analysis and the result analysis. The stage of qualitative analysis includes task 1 to 4, the stage of quantitative analysis includes task 5 to12, the results analysis includes task 13 to 15.

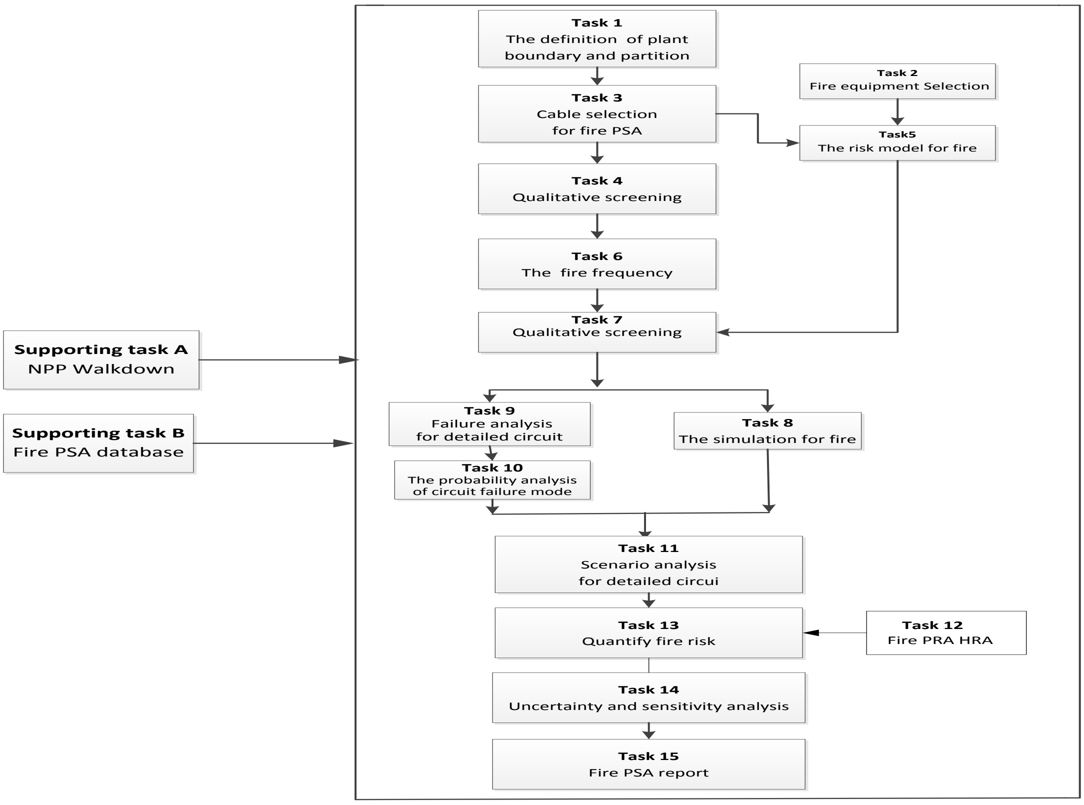

Fig.1 Flowchart of fire PRA.

4 Fire risk quantification

4.1 Qualitative analysis

Qualitative analysis begins with a liberal definition of plant boundary. The analysis boundary should be able to contain all locations which may contribute to fire risk for NPPs and then divided the global plant analysis boundary into discrete physical analysis units, namely, fire compartments. The equipment and cables which may be damaged by fire in each fire compartment were identified to support the analysis of the consequence of a given fire in the fire compartment. According to the consequences of fire, we performed a qualitative screening of the fire compartments and then screened out fire compartments where the fire risk is expected to be relatively low or nonexistent compared to others to confirm the fire compartments which should be done quantitative analysis.

All buildings and structures in NPPs were conducted screening analysis during the definition of plant boundaries. Those buildings and structures not leading the shutdown or impacting the mitigation function should be screened out. Reactor building, electrical building and other more than 20 buildings and structures are retained to be analyzed. Fire compartment division mainly refers to partitions in the fire protection design. Nearly 4000 equipment possibly damaged by fire was found out according to the NPP system design, electricity and equipment layout and walk-down *etc*. Fire equipment list was established. Based on the equipment list, we can identify the cables for power supply, control, display related to fire PRA equipment, and confirm the cables layout path and establish the fire cable lists.

During the qualitative screening, if the fire compartment simultaneously satisfied the following screening conditions and then screened out this compartment: 1) the fire compartment does not contain any fire PRA equipment and cables; 2) the fire in compartment will not lead to automatically shut down, or manually shut down required by emergency operating procedure, other power plant strategy, procedures or routine usage, or controlled shutdown due to the violation of the power plant operation technical specification.

4.2 Quantitative analysis

Quantitative analysis is the core content of fire PRA. For the fire compartment which is not screened out in the qualitative screening, we need find out ignition source and calculate ignition frequency, which will be as the input conditions to the fire risk model. Combined with fire simulation and circuit failure analysis, we establish the fire PRA model and complete the fire risk quantification.

4.2.1 The calculation of ignition frequency

The calculation of ignition frequency is the basis of fire scenario analysis and fire quantification.[8] The ignition frequency $\lambda_{IS,J}$ for the compartment is the sum of all ignition frequency in the fire compartment. The fire frequency is calculated by the following formula:

$$\lambda_{IS,J} = \lambda_{IS} W_L W_{IS,J,L} \quad (1)$$

Where:

λ_{IS} = plant-level fire frequency associated with ignition source IS;

W_L = location weighting factor associated with the ignition source;

$W_{IS,J,L}$ = ignition source weighting factor, reflecting the quantity weighting of the ignition source type present in fire compartment J of location L.

General ignition frequency derived from American fires events database, which can be obtained directly from NUREG/CR-6850 Table 5-1. Location weighting factor adjusts the frequencies for those situations where a common location are shared between multiple units. For example, because the auxiliary building serves two units, then 2.0 has been used for location weighting factor. Ignition source weighting factor represents the weight of ignition source number in the fire compartment.

The specific NPP is twin units, the calculation results of ignition frequency in the main area are shown in Table 1. Although most of the design of Unit 1 and Unit 2 are the same, there are some differences between the two units. For example, the routine and amount of cables in every room and the location of local control cabinets and junction boxes are not completely same.

Table 1 the ignition frequencies of plant locations.

Zone	Ignition frequency (reactor/year)	
	No. 1 power plant unit	No. 2 power plant unit
Reactor building	1.05E-02	1.04E-02
Electrical building	4.52E-02	4.15E-02
Fuel building	4.18E-03	4.25E-03
Connection building	1.08E-02	9.98E-03
Turbine building	7.75E-02	7.74E-02
Transformer area	1.72E-02	1.72E-02

4.2.2 The fire PRA model

Fire PRA model is based on the internal events of level I PRA model, the following technical elements of level I PRA model are included in the fire PRA model: initiating event analysis, accident sequence analysis, systems analysis, human reliability analysis and data analysis. In the analysis of each element, specific analysis or internal event model was modified due to the particularity of fire incidents, and then fire PRA model is established. These elements in Fire PRA model are handled as follows:

(1) Initiating event analysis: The initial events caused by the fire are analyzed based on fire equipment and cables affected by fire. Assume equipment and cables of fire PRA in compartment are all damaged after fire, the effects of damaged equipment and cables for NPP are analyzed to confirm the initial events caused by fire in the fire compartment.
(2) Accident sequence analysis: Automatic response and human response are simulated through the event tree after the initial incidents with reference to existing accident procedure. The main accident sequence adopted in this analysis are as follows: loss of all hot trap, loss of off-site power, loss of main feed water, general transient, loss of DC power, loss of compressed air, new sequence of fire events in main control room and sequence of fire events in equipment cooling water pump room.
(3) Systems analysis: Equipment has specific failure modes in the fire. The probability of failure may change due to the effect of fire, even as same as random failure modes in the internal events of PRA. So the modeling PRA software RiskSpectrum is applied to modify or supplement fault tree through the exchange event function, the failure of equipment and cables is reflected in the model.
(4) Human reliability analysis：The fire may cause equipment disoperation and instrument error display. Operator may have more pressure under the fire, which will affect their actions. Human reliability analysis mainly refers to the recommended method of fire HRA guideline (NUREG - 1921),[9] which include three phases: screening value, scoping and detailed analysis. During the human reliability analysis, we need change the performance shaping factor (PSF) to reflect the fire influence.
(5) Data analysis: Fire PRA model is established based on the internal event model and results of data analysis derived from internal events. Specific data in fire PRA includes fire frequency data, equipment failure data caused by fire and human reliability data after fire.

After fire PRA model was established, quantitative calculation was conducted under the conservative assumptions. The conservative assumptions that any fire source in the fire compartment ignites, which can lead to all the equipment and cables in the compartment occur the worst failure mode, to obtain the preliminary quantitative results of the fire compartment.

4.2.3 Quantitative screening

In the initial fire PRA model, preliminary quantitative fire risk was based on the conservative assumptions, which cannot actually reflect the fire risk for NPPs. Preliminary quantitative results should be selected by the appropriate selection criteria. A detailed analysis should be performed for risk-significant compartment to assess the risk of the fire compartment more accurately. The aim is to find out the risk contribution. Quantitative screening can assure that fire risk in the key areas should be analyzed in detail and all screened out fire compartments have relatively small cumulative risk to CDF.

The quantitative screening criterion recommended in NUREG/CR-6850 is that “the total contribution to CDF for the fire compartments screened out is less than 10% of the internal events CDF.” The CDF of internal events PRA of this NPP is 7.20E-06/reactor×year. 5.00E-08 /reactor×year was selected as the screening criteria. Sum of CDFs for all screened out fire compartments are less than 6.00E-07/reactor×year, so 5.00E-08 /reactor×year is a suitable screening criteria.

4.2.4 Detailed analysis

All the component and cables will be damaged if any fire source ignited in the preliminary quantitative assumptions. But in fact, fire has the developing and spreading process. Some fire source ignition may only affect the limited range and not cause all the component and cables damaged in the fire compartment. So we need simulate the fire occurrence and development through the fire modeling, which can estimate realistically the possible affected scope. In the selection fire PRA cables, the equipment circuit failure analysis is not in deep analysis, only the cables associated with equipment are listed. The failure of the cable not always leads to adverse failure modes, and the detailed analysis for the circuit failure are needed to confirm the actual impact of the cables to the component. Since the detailed analysis need spend a lot of manpower and material resources, only the fire compartment with higher risk will be conducted.

During the analysis, the circuit failure analysis for power, control and measuring cables are conducted in the motor circuit, electric valve circuit, distribution circuit and power circuit. That can confirm the response of the component to cable/circuit failure modes and screen out the cables that cannot affect the component to carry out function. Both the ground short circuit and hot short circuit failure modes are analyzed; the conservative thinking that probability of failure mode is 1. [10] We adapt FDTs and CFAST method to develop fire modeling for fire compartment. The main control room was modeled by CFAST.[11] The double-layer model was adopted to simulate the fire occurrence and development process in the main control room. For the other fire compartments, we adopt the calculation form based on the empirical formula of FDTs [12] and analyze whether the important target of fire PRA will be damaged by specific fire ignition.

4.3 The analysis results

4.3.1 The quantitative results

The definition of fire scenario is based on the results of the detailed analysis. The analysis of fire compartment level from the initial model was refined into the fire scenario level and fire PSA model. Finally quantitative results of fire PSA are obtained.

We select more than 200 fire scenarios for 2 units with over 100 fire compartments to carry out the quantitative analysis, eventually obtaining the CDF caused by fire. For example, the CDF point estimation of a unit is 4.03E-06/reactor×year. The quantitative results of the top five fire scenario and structures are respectively given in Table 2 and Table 3.

From the above analysis, the dominant fire scenarios are operator workstation fire in main control room, A# cable room fire in electrical building, low and medium voltage distribution room fire, DC and continual AC distribution room fire, A# cold water pump fire compartment fire. Important fire compartments of fire risk are the main control room, the cable of series A, accumulator and distributor rooms. Electrical building and main control room are the main contribution to internal fire CDF, reach about 30%. In addition, the analysis the conventional island includes turbine building, master-assistant transformer and substation, *etc.* which can result in the loss of main feed water, loss of off-site power and higher frequency of ignition, so the fire risk slightly higher than the nuclear auxiliary building.

Table 2 Quantitative results of fire scenarios.

Number	Fire scenario analysis codes	Fire scenario description	CDF (reactor×year)	The percentage of contribution
1	1MCR-S1	Operator workstation fire in Room 1L710	9.12E-07	24.80%
2	1MCR-S2	Other ignition fire except operator workstation in Room L710	2.66E-07	6.55%
3	1SFSL0380A-S8	Room 1L404(A# low voltage medium voltage distribution room fire)	1.88E-07	4.66%
4	1SFSN0287A	1SFSN0287A fire compartment(1#A RRI pump fire)	1.65E-07	4.10%
5	1SFSL0380A-S3	Room 1L304(A# cable room fire) transient fire	1.61E-07	3.99%

Table 3 Quantitative results of the buildings.

Number	The name of plant and structures	CDF(reactor×year)	The percentage of contribution
1	Main control room	1.40E-06	34.46%
2	Electrical building	1.22E-06	30.28%
3	Structures of conventional island	4.37E-07	10.88%
4	Nuclear auxiliary building	4.18E-07	10.36%
5	Connection building	1.70E-07	4.24%
6	Architectural structures of BOP	1.49E-07	3.73%
7	Reactor building	1.18E-07	2.95%
8	Fuel building	1.11E-07	2.70%
9	Emergency diesel generator building	1.61E-08	0.43%

Table 4 Sensitivity analysis results.

comparative item	Original CDF value	New CDF value	Occupation rate
Probability of human error increase10times	4.03E-6/reactor×year	5.12E-6/reactor×year	increased by 27.2%
Probability of human error reduce 10 times	4.03E-6/reactor×year	1.45E-6/reactor×year	reduced by 64.0%

Table 5 Sensitivity analysis results for screening criteria.

screening value	Original CDF value	New CDF value	percent change
1.0E-07/reactor-year	4.03E-6	4.35E-06	increased by 8.16%
2.0E-08/reactor-year	4.03E-6	3.90E-06	reduced by 3.23%

4.3.2 Sensitivity analysis

The potential correlation of sensitivity between equipment failure and human failure to CDF is identified. Those modelling assumptions, criteria of success and sensitivity have the potentially high impact on the results, including sensitivity analysis of initiating events, basic events, human error events and reliability parameters. The following is sensitivity analysis specific to parameter selection and so on.

(1) human reliability sensitivity analysis

In fire PRA model, the top 20 biggest human errors with the highest sensitivity, including 16 human error events after the initial event. Among them, the human error event that "shift engineer failed to announce to evacuate from main control room" has the largest sensitivity to the overall fire CDF. It is also a specific human failure event in fire PRA. The sensitivity analysis results for the human error event that "shift engineer failed to announce to evacuate from main control room" shown in Table 4.

We can see from the above analysis, the shift engineers announce to evacuate and personnel extinguishment have the bigger influence on results. The two events are dependent on the guidance of fire-fighting action card in the main control room. But, at present, the fire fight actions and evacuation procedures of specific circumstances in the fire-fighting action card are not explicitly defined, only depends on the individual judgment of shift engineer. It is suggested that fire-fighting action and the execution standard of evacuation procedures in main control room should be clear in fire-fighting action card. Fire drill of main control room is enhanced to ensure the operators to perform timely the correct action under the fire scenario.

(2) sensitivity analysis of quantitative screening value

The purpose of quantitative screening criteria is to ensure that the cumulative risk contribution for screening out the fire compartments should be relatively small. The selection of criteria determines amount of analysis sequence, the reserved fire compartments by quantitative screening should develop the further fire simulation and circuit failure analysis. Therefore, it is necessary to have sensitivity analysis for screening criteria. Screening criteria used in this paper is "all the sum contribution to CDF with screening out fire compartments that should be less than 10% of the total internal events CDF". According to the PRA results of the internal events, 5.0E-08/reactor×year was selected as quantitative screening value of fire compartment. The screening value was compared and analyzed through mitigation and amplification of values, the results was shown in Table 5.

Table 5 shows that when the screening value is increased twice of original value, the CDF grows significantly. When the screening value is reduced to half of the original value, CDF reduce less. Through the above results, when the screening value increasing, the amount of fire compartments which need a detailed analysis is decreased, but CDF increased significantly, which cannot reflect the actual situation of fire risk. When the value reduces, the number of fire compartments increase, the workload is too large that will spend a lot of manpower and material resources, but CDF with little decrease, which shows the risk contribution of fire compartments have little impact on the total results, there is no need to have a detailed analysis. Therefore, to adopt the reasonable quantitative screening criteria can balance the workload and maximally have risk analysis, and give the important risk contribution and insights.

5 Conclusion

Through fire PRA analysis, weak links of the existing structures, systems, component can be identified. We can find out the defects of system, component, fire protection design and important human actions and so on. It is helpful for the NPPs to improve ability of hazards response and safety level through reconstruction or improvement.

Fire PRA has a lot of uncertainty due to selection of parameters and methods. Therefore, having the sensitivity analysis is to find important items. The results of this analysis can be seen as follows:

- Electrical building and the main control room are the more obvious fire risk for internal fire PRA. A large number of electrical equipment and cables such as switchboard layout inside the electrical building, and fire frequency is higher. There are so many control and surveillance activities in the main control room. After a fire, control panel indoor all damage or operator can't stay and loss the monitoring ability of units.
- Human action is significant in the analysis, especially the fire scenario that shift engineer announce to evacuate and personnel put out fire have a greatly influence on results. There is a big uncertainty that only depending on the judgment of operator and shift engineer. It is suggested the fire-fighting actions and standard evacuation procedures of specific circumstances should be defined in the fire-fighting action card to ensure the operators not to lose the monitoring of power plant.
- Qualitative and quantitative screening is significant characteristics for hazard PRA. Reasonable selection criterion is helpful to reduce the workload and have maximum risk insights. From the sensitivity analysis, we can see that it is reasonable to adopt 10% of PRA results as the quantitative screening criteria.

References

[1] NI M., GONG Y., and XIAO J., *et al.:* Dission on the Method of Detail Fire Scenario in Fire PSA of Nuclear Power Plant. Nuclear Safety [J].2015.

[2] EPRI. Fire-induced vulnerability evaluation (FIVE), EPRI TR-100370 [R]. 1992.

[3] EPRI. Fire PRA implementation guide, EPRI TR-105928 [R]. 1995.

[4] IAEA. Safety Reports Series No.10, Treatment of internal fires in probabilistic safety assessment for nuclear power plants [R]. 1998.

[5] EPRI/NRC. Fire PRA Methodology for Nuclear Power Facilities. NUREG/CR-6850 Vol.2 [R]. 2005.

[6] YU X.L., ZHENG X.Y., and ZHAO B.: The State of Art of Internal Fire PSA in Nuclear Power Plants [J]. Nuclear Safety, 2010.

[7] HU X.M.: Analysis of Fire Frequency in Nuclear Power Plants [J], 2010.

[8] ASME/ANS. ASME/ANS RA-SA-2009 Standard for level 1/large early release frequency probabilistic risk assessment for nuclear power plant applications. 2009

[9] NUREG/CR-1921, Fire Human Reliability Analysis Guidelines Final Report. 2012.

[10] NUREG/CR-7150, Joint Assessment of Cable Damage and Quantification of Effects from Fire. 2012.

[11] CFAST-Consolidated Model of Fire Growth and Smoke Transport (Version 6) Technical Reference Guide. 2005.

[12] NUREG-1805, Fire Dynamics Tools (FDTS) Quantitative Fire Hazard Analysis Methods for the U. S. Nuclear Regulatory Commission Fire Protection Inspection Program, 2004.