Fire Probabilistic Risk Assessment of BWR-6 Mark-III Nuclear Power Plant for the Pre-Defuel Phase in the Decommissioning Transition Stage

Yu-Ting Lin, Yu-Min Chen, Yi-Cheng Tian, Tsu-Mu Kao

National Atomic Research Institute (NARI)
No. 1000, Wenhua Rd., Longtan Dist., Taoyuan City 325207, Taiwan (R.O.C.)
anita.lin@nari.org.tw; ymchen@nari.org.tw; leontian@nari.org.tw; tmkao@nari.org.tw

ABSTRACT

This paper evaluates the fire probabilistic risk during the extended pre-defuel phase of a boiling water reactor (BWR) nuclear power plant in the decommissioning transition stage, that is, under the condition when the last batch of spent nuclear fuel assemblies is still in the Reactor Pressure Vessel (RPV). Internal fires are recognized as a significant contributor to overall risk in nuclear power plants. Generally speaking, the results of fire probabilistic risk assessment (FPRA) are often influenced by factors such as research schedules, manpower, expenditures, and the subjective judgment of analysts, which may potentially lead to biased risk rankings. This study introduces a systematic, computational approach to simulate and quantify fire scenarios objectively. This method ensures consistent risk evaluation and provides a foundation for future in-depth research. Using the WinNUPRA software package, a fire scenario analysis was performed based on the initial state of the plant 178 days after permanent shutdown. Two configurations are examined: the reactor core and the spent fuel pool (SFP). The scope of this study focuses on the fire scenario combination of all ignition sources and targets in a single compartment. A case study on a BWR-6 Mark-III plant demonstrates that the fuel uncovery frequency (FUF) for the core configuration accounts for 97.04% of the total risk, while the SFP contributes 2.96%. These results show that the reactor core is the primary contributor to the risk of fire during the pre-defuel phase.

Keywords: FPRA, Decommission, Pre-defuel

I. Introduction

The pre-defuel (PD) phase is the transition phase before the decontamination and demolition stages. The PD phase is similar to the low power and shutdown (LPSD) phase. The focus of this study is limited to the accident sequences and risk profiles of fuel damage in the reactor and the spent fuel pool (SFP) caused by fire events under pre-defuel conditions.

This study involves two models, one with the fuel from the last cycle in the reactor pressure vessel and the other with the total spent fuel in the spent fuel pool above the fuel storage building. The reference type of this nuclear power plant is twin/BWR-6/Mark-III. During normal operation of the PD phase (18 months), there is no movement of fuel, and the transfer pipeline is isolated by manual valves (i.e., the upper pool is separated from the spent fuel pool). Methodology is illustrated in section II, details and results of quantification are discussed in section III, and a summary is made in section IV.

II. Methodology

This study refers to the FPRA methodology (NUREG/CR-6850) [1][2] and adjusts the in-plant fire assessment used in the decommissioning transition stage analysis with 11 tasks, as shown in Figure 1. The contents of each task are described below.

II.A. Plant boundary & partitioning

This section covers task 1. The global plant analysis boundary is defined in a broadly inclusive manner. Unimportant areas within this boundary will be readily identified and eliminated from further analysis during the early screening task (e.g., Task 4). Before the early screening work, unimportant areas are screened out according to the screening criteria. For example, administrative office buildings or warehouses within the protection boundary can be screened out. The area does not contain relevant equipment that will interrupt the cooling of the reactor core or spent fuel pool, and it is an independent building and has no fire spread path. These areas do not require further analysis.

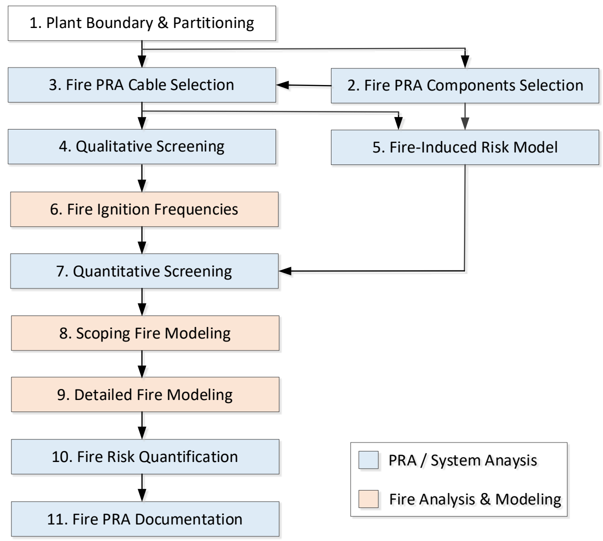

FIGURE 1. Fire Probabilistic Risk Assessment Process

II.B. Selection and qualitative screening of fire components and cables

This section covers tasks 2, 3, and 4. In the fire probabilistic risk analysis, if the subject of fire damage is "equipment related to the fuel cooling interruption of the reactor core or the spent fuel pool," it must be included in the fire PRA model. The subject included in the fire PRA contains fire PRA components and related cables. Based on the functions of system components and the actions in an emergency, as well as the fire PRA component selection principles, include all system components that may be affected by fire in the component selection list. To establish a fire PRA component list, it is necessary to confirm the cable information and the location of each component. Information about cables connected to each component and the fire compartments they pass through is used to analyze the occurrence of fires in each fire compartment and the equipment inside. From the cables connected to each component to the fire compartments these cables pass through, all this information is used to analyze the occurrence of fires in each fire compartment and the equipment inside, ultimately evaluating the risk of fire.

The qualitative screening of fire zones is to reduce the number of fire zones and fire scenarios that need to be analyzed in the quantitative screening analysis step. The screening criteria of qualitative screening are:

(1) Code I: Fire compartments contain important safety equipment and must be retained for quantitative screening analysis;
(2) Code N: Even if all equipment in the fire compartment is damaged by fire, equipment not related to maintaining fuel cooling for the reactor core or the spent fuel pool can be directly evaluated using the screening criterion between compartments. If there is no spread to another compartment, it can be screened out directly.;
(3) Code U: Fire compartments containing cables related to post-fire PRA equipment are retained for quantitative screening analysis.

The above three items are also called fire impact codes. For those belonging to code N and with no effective fire spread path, it can be directly screened out.

II.C. Fire ignition frequency

This section addresses Task 6. The fire ignition frequency analysis integrates the NUREG/CR-6850 [2] fire methodology and NUREG-2169 [3] low-power and shutdown data to identify fire sources across the entire plant. The analysis categorizes fire sources into 37 fire bins, including batteries, reactor pumps, transformers, and others. Frequencies are assigned to individual devices or fire compartments based on device counts or parameter scores for various bins. Additionally, the methodology of FAQ 12-0064 [4] is applied to analyze fire type bins and frequencies of temporary fire sources. The total fire frequency for a fire compartment is the sum of the frequencies assigned to each fire bin within that compartment.

II.D. Fire risk model establishment and quantitative screening analysis

This section covers Tasks 5 and 7. The first step in establishing the fire Probabilistic Risk Assessment (PRA) power plant response model is to identify initiating events caused by fire damage in a fire compartment, compared to the internal event PRA model. Initiating events are determined based on the equipment in the fire compartment, the cables passing through it, and a comprehensive fire cable database. In the preliminary quantitative screening stage, it is assumed that all equipment and cables in the fire compartment sustain the most severe fire damage, resulting in equipment failure, cable failure, and the simultaneous occurrence of all failure modes. If the failure of equipment and cables in a compartment does not lead to the loss of the decay heat removal function, no initiating event determination is required for that compartment. When a fire compartment may cause multiple initiating events, all possible combinations of initiating events and equipment failures are quantified, and the combination with the highest Conditional Fuel Uncover Probability (CFUP) is selected as the initiating event. If maximum CFUP values are equal, the selection is based on the CFUP ranking of each initiating event under no-equipment-damage conditions.

Quantitative screening analysis uses the results of Task 6 ("Fire Ignition Frequencies") and Task 5 ("Fire-Induced Risk Model") as inputs. Screening criteria ensure that the cumulative risk of screened-out fire compartments is minimal. The criteria for a single compartment follow the PRA standard (ASME/ANS RA-Sa-2013) [5], issued by the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) in 2013, specifically Table 4-2.8-4 for Capability Category II. The Fire Uncover Frequency (FUF) of screened compartments is sorted and accumulated from smallest to largest. Compartments contributing to a cumulative FUF exceeding 10% are retained for detailed analysis. The risk of fire compartments that are quantitatively screened out will adopt their original value and be combined with detailed results to estimate the total fire risk.

II.E. Fire Definition and Simulation

Tasks 8 and 9 address fire definition and detailed fire simulation, respectively. Task 8 defines the severity factors (SF) for fire bins in simulations, while Task 9 conducts detailed fire simulations. For each unscreened fire source in a fire compartment, relevant fire scenarios are identified, and detailed simulations are performed to calculate the probability of a fire not being extinguished.

The purpose of fire definition and simulation is to use fire simulation tools to accurately model the combustion scale of fire sources, estimate the likelihood of fire spreading to target objects, and evaluate fire scenario probabilities. Key parameters, such as burning time, fire protection system response, and non-extinguishing probability, are incorporated into the quantitative fire risk calculation. This task builds on the fire frequency analysis in Section II.C, using fire source and compartment data to conduct defined fire simulations and detailed site surveys. The process and results of single-compartment fire simulations are performed for compartments retained after the quantitative screening analysis in Task 8.

Automatic suppression system failure rates are based on NUREG/CR-4840 [6]. The unavailability of water and carbon dioxide systems is 0.04, and the unavailability of the Halon system is 0.06, representing the failure factors for automatic fire protection systems.

II.F. Quantitative analysis of fire risk

This section addresses Task 10, focusing on fire prevention compartments retained from the quantitative screening analysis in Task 7 and conducting detailed fire scenario simulations and fire risk quantification in Task 9. Fire scenario selection is the primary focus of detailed fire simulations in Task 9. During fire scenario analysis, the relationship between each fire source and the target is examined individually to calculate the probability of fuel exposure. The frequency of fuel exposure is obtained by multiplying the fire frequency, severity factor, non-extinguishing probability, and fuel exposure probability.

III. Nuclear power plant during the decommissioning transition period

The fire PRA methodology discussed in Section II is applied to the BWR-6 Mark-III nuclear power plant in the decommissioning transition period. During decommissioning, fuel is transferred from the reactor core to the spent fuel pool. The transition period occurs when the spent fuel pool lacks sufficient space for all fuel, requiring some fuel to remain in the core. This section demonstrates an application case following the fire PRA process.

III.A. Plant boundary & partitioning

The analysis boundary for the power plant was established by referencing the structural overview and the Final Safety Analysis Report (FSAR). Of the 67 buildings in the BWR-6 Mark-III nuclear power plant included in the analysis boundary, 61 were preliminarily screened out. Six buildings require qualitative screening and analysis of their fire compartments: the reactor building, reactor auxiliary building, turbine building, control building, fuel building, and emergency seawater pump room.

III.B. Selection and qualitative screening of fire components and cables

After establishing the fire PRA component list using the internal event safety assessment model for the decommissioning transition phase of the BWR-6 Mark-III nuclear power plant, the location of each component, its connected cables, and the fire compartments these cables traverse are confirmed. This enables analysis of fire risks for equipment and cables and their post-fire impacts. A correlation database integrating components, cables, and fire compartments is created, followed by qualitative screening analysis. The fuel building is used as an example to illustrate the results of preliminary screening.

The fuel building contains 24 fire compartments. Based on the floor plan of critical equipment, fire countermeasure procedures, and analysis data from the in-plant incident safety assessment model during the decommissioning transition phase, the equipment in each fire prevention zone was evaluated. Qualitative screening determines whether a compartment contains "equipment that may interrupt cooling of the reactor core or spent fuel pool" or a "potential fire source":

(1) Directly screened out: Three fire compartments lack equipment that could interrupt fuel cooling in the reactor core or spent fuel pool and have no effective fire propagation path, allowing them to be screened out..
(2) Retained for quantitative screening analysis:
   a. Four fire compartments contain equipment that could interrupt cooling of the reactor core or spent fuel pool ("I" category) but have no effective propagation path.
   b. Five fire compartments contain "I" category equipment and have an effective propagation path.
   c. Seven fire compartments contain cables related to the safety shutdown system ("U" category) but have no effective propagation path.
   d. Three fire compartments contain "U" category cables and have an effective propagation path.

In total, 19 fire compartments require quantitative analysis.

III.C. Fire ignition frequency analysis

During the calculation of fire frequency, the equipment in each fire compartment must be classified according to different fire bins. After the equipment in all fire compartments has been inventoried, the proportion of a single piece of equipment in the fire bin must be calculated, along with the fire frequency of a single bin for that equipment. The final step is to sum the frequencies of each fire bin within a single fire compartment to obtain the fire frequency of the fire compartment.

During the fire frequency analysis, fire bins are divided into three categories based on their characteristics: (I) Fire bins with physical equipment that can be directly counted. (II) Fire bins that cannot be directly counted; weights are assigned based on relevant data. (III) Fire bins with no counting data available.

Category I can be counted directly. During the fire frequency analysis process, the number of equipment must be counted in detail. In addition to collecting relevant supporting information from the power plant, power plant staff are also asked to assist with interpretation during the process, particularly for confirming new or removed equipment. Special counting methods for some fire bins, as classified in NUREG/CR-6850 [1][2], are applied and explained as follows:

(1) Bin 15 counts non-high-energy arc electrical cabinets by their vertical segments.
(2) Bin 16.a and Bin 16.b: For detailed calculation of electrical cabinet panels (e.g., motor control centers, switchgear, load centers), equipment is counted according to vertical segments.
(3) Bin 16.1: Bus ducts are counted based on the number of joints in the metal casing of the bus duct.

Based on the counting results, the proportion of each piece of equipment in a fire bin is determined. For example, if the total number of pumps in fire bin 21 across the entire plant is 153.5, each pump accounts for 1/153.5 of the fire frequency. The fire frequency of bin 21 is 2.72E-02/yr; when multiplied by 1/153.5, this equals 1.77E-04/yr, which is the fire frequency for one pump. The calculation method for other fire bins follows the same approach.

The ignition frequency of Category II temporary fire sources is calculated per fire compartment. Each room is assigned a counting weight and divided into three counting methods:

(1) Fires caused by welding or cutting for temporary fire sources (bins 6, 24, and 36): The influencing factor is the number of high-temperature maintenance operations in each fire compartment. The reference material is the fireworks order of the analyzed plant.

(2) Cables on fire caused by welding or cutting (bins 5, 11, and 31): In addition to high-temperature maintenance operations, influencing factors also include the weight of cables in the fire compartment. The reference for cable weight is the Pre-Fire Procedure.
(3) Temporary fire sources (bins 7, 25, and 37): Influencing factors include general electromechanical maintenance operations, the number of on-site workers, and stored materials. The reference materials are the general electromechanical maintenance work order information and the Abnormality and Emergency Operation Procedure Book.

Additionally, due to significant difficulties in counting cables and junction boxes in fire compartments, the fire frequency is directly allocated based on the proportion of a single fire compartment's cables relative to the entire plant's cables, with reference to the cable weight in the Fire Countermeasures Procedure Book.

Category III uncounted information includes:
(1) Bin 2 (reactor): Applicable only to PWR power plants and thus not included in this study.
(2) Bin 30 (boilers): Located outside the turbine building and differing from the NUREG/CR-6850 [1][2] combustion bin counting definition, so they are excluded from the calculation.

The fire source inventory is one of the key preparatory tasks for the on-site survey, and the survey results have been incorporated into the fire bin count. The NUREG/CR-6850 [1][2] methodology was used to count the 37 fire bins across nearly 400 fire compartments in the analyzed plant. After determining the total number of fire bins in the entire plant, the fire frequency from NUREG-2169 (low power and shutdown data) [3] is divided by the total number of fire bins in the plant, multiplied by the number of fire sources in each fire compartment, and then the fire frequencies of each fire source in each compartment are summed to obtain the fire frequency of the fire compartment.

III.D. Fire risk model establishment and quantitative screening analysis

For fire compartments retained after qualitative screening, a preliminary quantitative analysis is conducted using conservative assumptions. Fire compartments whose results meet the quantitative screening conditions are excluded, further reducing the workload of subsequent detailed analysis.

Quantitative screening of fires during the decommissioning transition period of the BWR-6 Mark-III nuclear power plant is performed. The screening results for initiating events that may lead to the risk of fuel uncovery in the reactor core or spent fuel pool are explained below and summarized in Table 1:
(1) AAA-LOR (Loss of RHR): When critical components related to the A and B loops of the RHR (Residual Heat Removal) system are lost simultaneously, the initiating event is classified as AAA-LOR.
(2) AAA-TCH (Loss of Normal Chilled Water): When critical components of the normal chilled water system fail, the event is designated as AAA-TCH.
(3) AAA-TCE (Loss of Essential Service Water): When critical components of the essential service water system fail, the event is designated as AAA-TCE.
(4) AAA-TA3 (Loss of 4.16 kV Bus 1A3): When critical components related to the 4.16 kV bus 1A3 fail, the event is classified as AAA-TA3.
(5) AAA-TA4 (Loss of 4.16 kV Bus 1A4): When critical components related to the 4.16 kV bus 1A4 fail, the event is classified as AAA-TA4.
(6) AAA-LOC (Loss of SFPCCS): The nuclear component closed cooling water (NCCCW) System provides the cooling water for the heat exchanger and pump cooler of the spent fuel pool cooling and cleanup system (SFPCCS). Therefore, when the critical components of the NCCCW or SFPCCS fail, the event is classified as AAA-LOC.
(7) NA Category: Initiating events not falling into categories 1 through 6 are classified as not causing an initiating event (NA category).

If the analysis identifies more than two simultaneous initiating events, the event with the most severe consequences is selected. For example, if AAA-LOR and AAA-TCE occur simultaneously, the event is classified as AAA-LOR. In the quantitative screening stage, the determination of the initiating event and risk quantification for a fire compartment assumes conservatively that all equipment within the compartment fails. Human factor basic events in the FPRA model are also conservatively used in the screening analysis values. If the quantitative results under these conservative assumptions still meet the screening conditions, the actual risk contribution is expected to be lower.

TABLE 1. The Screening Results and Classification of Initiating Events

Initiating Events PRAID	Description	Initiating events at the reactor core	Initiating events at the spent fuel pool
AAA-LOR	Loss of RHR	✓	
AAA-TA4	Loss of 4.16kV Bus 1A4	✓	
AAA-TA3	Loss of 4.16kV Bus 1A3	✓	
AAA-TCE	Loss of Essential Service Water	✓	
AAA-TCH	Loss of Normal Chilled Water	✓	
AAA-LOC	Loss of SFPCCS		✓

Quantitative analysis is divided into quantitative screening and detailed analysis. After qualitative screening, fire compartments containing critical equipment or cables proceed to quantitative screening. If critical equipment in a fire compartment catches fire, is damaged due to fire affecting other equipment in the compartment, or if the fire spreads to other compartments and causes the fuel uncovery frequency (FUF) sequence of the reactor core or spent fuel pool to exceed 10% of the FUF for internal events at the site, further detailed analysis is required. The risks of each compartment are ranked and accumulated from smallest to largest. The sorting threshold is reached when the cumulative value exceeds 10% of the FUF for internal events. The quantitative screening values for FUF are 5.01E-09/yr for the reactor core and 3.26E-09/yr for the spent fuel pool, with a total quantitative screening FUF of 8.27E-09/yr. Consequently, 41 compartments require detailed analysis. The quantitative screening results, using the fuel building as an example, are detailed below.

In the fuel building, 19 fire compartments require quantitative screening analysis. The screening results are as follows:

(1) Cumulative Value Exceeding 10% of Internal Event FUF: No fire compartments exceed this threshold, and thus, further detailed analysis is required for zero compartments.

(2) Cumulative Value Not Exceeding 10% of Internal Event FUF: Nineteen fire compartments meet this criterion, requiring no detailed analysis, and their values can be directly adopted.

III.E. Fire scenarios and quantitative analysis

Fire scenario selection is the primary task of detailed fire simulation. Single-compartment fire scenarios are based on quantitative screening of retained fire sources within the compartment. The risk of each fire scenario is calculated by evaluating severity factors (SF), non-suppression probability (Pns), and fire damage consequences. This study proposes a method based on the NUREG/CR-6850 methodology [1][2] and a systematic process [4], ensuring a consistent approach to fire scenario development and minimizing risk bias across compartments. The following sections discuss the fire scenario analysis for ignition sources: equipment, cables, temporary ignition sources, and induced cables (IC).

III.E.1. Equipment fire scenarios

These scenarios involve fires occurring in equipment and explore their potential impact on power plant risks. Fire bins included in equipment fire scenarios are: Bin 1 (Batteries), Bin 8 (Diesel Generators), Bin 9 (Air Compressors), Bin 10 (Battery Chargers), Bin 14 (Electric Motors), Bin 15 (Non-High Energy Arcing Faults (HEAF) Electrical Cabinets), Bin 16.a (480V–1000V HEAF Electrical Cabinets), Bin 16.b (>1000V HEAF Electrical Cabinets), Bin 16.1 (Bus Ducts), Bin 16.2 (Isolated Phase Bus), Bin 19 (Other Hydrogen Fires), Bin 21 (Pumps), Bin 22 (RPS MG Sets), Bin 23 (Transformers), Bin 26 (Ventilation Subsystems), Bin 32 (Main Feedwater Pumps), Bin 33 (Turbine/Generator (T/G) Exciter), Bin 34 (T/G Hydrogen), and Bin 35 (T/G Oil).

Analysis of equipment-based fire scenarios requires examining the relationship between each ignition source and target object individually, using NUREG/CR-6850 [2] "TASK 8 Scoping Fire Modeling" and "TASK 11 Detailed Fire Modeling" methodologies, and the Fire Dynamics Tools (FDTs) (Version 1805.1) spreadsheet [7] provided by the U.S. NRC. Fire damage time is estimated based on the damage mechanism of the target object. By repeatedly comparing results with on-site investigation data, each equipment fire scenario and its corresponding damaged equipment are accurately determined, and the severity factor and non-suppression probability are calculated. Using the internal event PRA model [8], the PRA basic event (PRAID) corresponding to the damaged equipment is set for each scenario, and the Core Damage Frequency Uncovery Probability (CFUP) is calculated. The FUF for the reactor core and spent fuel pool in equipment fire scenarios are 2.70E-07/yr and 4.87E-09/yr, respectively, with a total FUF of 2.75E-07/yr.

III.E.2. Cable fire scenarios

These scenarios begin with a fire in a cable and extend to potential impacts on power plant risks. Since cable materials significantly influence fire scenario development, the cable material type must be confirmed before analysis. Cable coatings are categorized as thermoset plastic (TS) or thermoplastic polyester (TP), each exhibiting distinct behaviors in fire environments. When burned, thermoset plastics form a carbonized layer that inhibits combustion, while thermoplastic plastics melt due to low melting points, producing droplets that create new ignition points and accelerate flame spread. Consequently, thermoset-coated cables burn only themselves, whereas thermoplastic-coated cables may spread fire to other cables. In developing cable fire scenarios, TS and TP are critical parameters, classified based on cable insulation material data from the power plant database. If insulation material data is unavailable, a conservative assumption of thermoplastic is applied. The FUF for the reactor core and spent fuel pool in cable fire scenarios are 1.23E-08/yr and 7.09E-10/yr, respectively, with a total FUF of 1.30E-08/yr.

III.E.3. Temporary fire source (including Induced Transient, IT) fire scenarios

Temporary fire sources, including those caused by welding and cutting, share the same heat release rate and influence area and are thus combined into temporary fire source (including IT) scenarios. Given the variety of transient ignition sources (e.g., liquid solvent fuels, solid fuels of differing types, sizes, and configurations), two assumptions are made: (1) a transient ignition source (including IT) burning within 0.9 m of equipment directly damages it, regardless of suppression success; and (2) fire spread scenarios consider secondary combustible propagation without accounting for the total heat release rate (HRR) of the transient source and secondary combustibles. The FUF for the reactor core and spent fuel pool in transient fire scenarios are 1.51E-07/yr and 4.47E-09/yr, respectively, with a total FUF of 1.55E-07/yr.

III.E.4. Induced Cable (IC) fire scenario

IC fire scenarios involve cable fires caused by welding and cutting. Since the fire originates externally, it may spread regardless of whether the cable is coated with TS or TP. It is conservatively assumed that a cable fire from welding or cutting immediately spreads to all cables in the affected cable tray. However, due to the presence of personnel during welding and cutting, the fire is unlikely to spread to other trays, limiting damage to a single tray. When quantifying IC risks in fire compartments, the frequency of occurrence is assumed to be evenly distributed across each cable tray. All IC fire incidents occurring within a compartment are assessed collectively, with the CFUP calculated for each tray and the results subsequently averaged. The FUF for the reactor core and spent fuel pool in IC fire scenarios are 7.54E-14/yr and 0.00E+00/yr, respectively, with a total FUF of 7.54E-14/yr.

III.E.5. Results

The PRA analysis results for fires during the decommissioning transition phase are presented in Tables 2 and 3. The total FUF is 4.44E-07/yr (Table 2), with equipment and transient fire scenarios as the primary risk contributors. The fire risk from quantitative screening of a single compartment is 8.27E-09/yr, while detailed analysis yields 4.43E-07/yr. Table 3 summarizes the overall fire risk quantification. Combining quantitative screening and detailed analysis results, the FUF caused by internal fire events is 4.38E-07/yr for the reactor core (97.03% of total FUF) and 1.34E-08/yr for the spent fuel pool (2.97% of total FUF). These results indicate that the reactor core is the primary risk contributor during the pre-defueling phase.

TABLE 2. Detailed Fire Risk Quantification Results of Single-Compartment Fire Scenarios

Item	Fire Scenario Analysis of Ignition Sources	Reactor Core-FUF (1/yr)	Spent Fuel Pool –FUF (1/yr)	FUF (1/yr)	FUF %
1	Equipment Fire Scenario	2.70E-07	4.87E-09	2.75E-07	62.08%
2	Cable Fire Scenario	1.23E-08	7.09E-10	1.30E-08	2.93%
3	Transient Fire Scenario	1.51E-07	4.47E-09	1.55E-07	34.99%
4	Induced Cable Fire Scenario	7.54E-14	0.00E+00	7.54E-14	0.00%
Total		4.34E-07	1.01E-08	4.44E-07	100.00%

TABLE 3. Overall Fire Risk Quantification Results for Single-Compartment Fire Scenarios

Item	Quantitative Screening (1/ yr)	Detailed Analysis (1/ yr)	Total FUF (1/yr)	FUF %
Reactor Core-FUF	5.01E-09	4.34E-07	4.38E-07	97.03%
Spent Fuel Pool -FUF	3.26E-09	1.01E-08	1.34E-08	2.97%
Total FUF (1/yr)	8.27E-09	4.43E-07	4.51E-07	100.00%

IV. Conclusions

This study adheres to the Fire PRA Methodology [2] developed by the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (RES) and the Electric Power Research Institute (EPRI). Preliminary qualitative screening identified six buildings requiring further analysis: the reactor building, reactor auxiliary building, turbine building, control building, fuel building, and emergency seawater pump room. Initiating events included in the quantification are AAA-LOR, AAA-LOC, AAA-TCH, AAA-TA3, and AAA-TA4. The results show that the reactor core, with a 97.04% contribution to FUF, is the primary fire risk contributor during the pre-defueling phase, while the spent fuel pool contributes a relatively minor 2.96%.

REFERENCES

[1] EPRI/NRC-RES, "Fire PRA Methodology for Nuclear Power Facilities: Volume 1, Volume 2." EPRI 1011989 and NUREG/CR-6850, 2005.

[2] EPRI/NRC-RES fire PRA methodology for nuclear power facilities: Supplement 1. EPRI 1011989 and NUREG/CR-6850, 2010.

[3] U.S. NRC, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database," EPRI 3002002936 and NUREG-2169, 2015.

[4] Pi-Lin Hsu, Chung-Kung Lo, "A systematic process for developing fire scenarios in risk assessment for nuclear power plants," Annals of Nuclear Energy (2021).

[5] ASME/ANS RA-Sb-2013 – Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, July 2013.

[6] M.P. Bohn, et al., "Procedures for the external event core damage frequency analysis for NUREG-1150", Sandia National Laboratories, NUREG/CR-4840, November 1990.

[7] U.S. NRC, "Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program - Supplement 1," NUREG-1805, July 2013.

[8] Yi-Cheng Tian, Chen-Che Hsu, Po-Jung Chiu, Tsu-Mu Kao, "Probabilistic Risk Assessment for the Transition Phase of Decommissioning of BWR-6 Mark-III Nuclear Power Plant," PSAM17&ASRAM2024.