CANDU Fire Probabilistic Risk Assessment

By

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Abstract

Advancement of fire risk analysis methods has led to a widespread development of detailed fire probabilistic risk assessments (PRA) at nuclear power plants. The Fire PRA assesses the possibility of a fire at critical plant locations and evaluates the fire damage. Fire PRA also evaluates the effect of the fire on safety-related cables and equipment. The scope of the Fire PRA is limited to demonstrating that the fire safe shutdown objectives and performance criteria are met. Hence, the Fire PRA is only used for plant areas where fires may have a potential impact on systems, structures, and components (SSCs) that are required to perform the fire safe shutdown functions.

Canadian Nuclear Power Plants (NPPs) use NUREG/CR-6850 or some portion of it in performing Fire PRAs. There are differences between Canadian and U.S. nuclear reactor types. The generic fire ignition frequencies provided in NUREG/CR-6850 reflect the experiences only of the U.S., and not Canada. There are also differences in systems, structures, and components when comparing Canadian nuclear reactors, which use pressurized heavy water, to U.S. reactors, which use light water. Differences are also found in the core of the reactors, which contain uranium fuel. CANDU nuclear reactors have a number of inherent safety features that differentiate them from light water reactors (LWRs), while light water reactors have other systems that do not exist in CANDU plants. Consequently, fire events that are related with these types of systems in LWR plants should not be considered in CANDU plants.

A CANDU Fire Database was developed by the Canadian Nuclear Safety Commission (CNSC) to collect and maintain data of all CANDU related fires. There are 19 nuclear power plants operating in Canada today, with 76 fires reported from 1981 to

2017. There was a spike in the number of fires between 1999 and 2005, which may be attributed to the change of reporting criteria between 1981 and 2017. From 1981 to 1996, the fire reporting criteria for all CANDU licensees were listed in their licence condition handbooks, and from 1996 to 2003, the fire reporting criteria followed "R-99: Reporting Requirements for Operating Nuclear Power Facilities". From 2003 to 2014, the fire reporting criteria followed "S-99: Reporting Requirements for Operating Nuclear Power Plants". From 2014 to present day, the fire reporting criteria have been following "REGDOC-3.1.1: Reporting Requirements for Nuclear Power Plants (NPPs)". There is a clear relationship between the increased numbers of fires reported since the transition from listing the fire reporting criteria in regulatory documents from the licence condition handbook in 1996.

The main objective of this thesis is to develop a Fire PRA for Canadian CANDU nuclear reactors. For this, a fuel load survey of all 1,230 Fire Safe Shutdown System (FSSA) rooms in CANDU reactors in Canada was carried out. A general fire zone list for all sites was developed in order to group fire zones with similar functions. The result was the identification of 38 general fire zones. The fuel load survey, which was carried out using the NFPA 557 combination method, found that the average fuel density for the 1,230 general fire zones was 170.1 MJ/m^2, and electric faults were the most likely ignition source. The results of the fuel load survey were used to group fire zones based on their areas, heights and combustible loads and to produce a list of critical fire scenarios. The computer program, Fire Dynamics Simulator (FDS) was used to simulate these critical fire scenarios and the associated consequences on FSSA cables. The fire scenarios' simulation results were used to create a qualitative screening method (decision

tree) for CANDU reactors, which is an essential step in the CANDU Fire PRA. A CANDU Fire PRA methodology for CANDU reactors was developed, and two fire zones were selected to demonstrate the use of the CANDU Fire PRA methodology. In addition, High Energy Arc Fault (HEAF) risks in CANDU reactors were examined and analyzed, and recommendations were given to mitigate the risks and consequences of any potential HEAF fire events.

Acknowledgments

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Prior Publications

Much of this thesis was previously published in conference and journal papers. Below is the list of these publications:

1) H. Shalabi and G. Hadjisophocleous: "Qualitative Analysis in CANDU Fire Probabilistic Risk Assessment (PRA)", CNL Nuclear Review, 2020 (Accepted and to be published)
2) H. Shalabi and G. Hadjisophocleous: "Decision Tree for FSSA rooms in CANDU", Nuclear Engineering and Design, 6 March 2020
3) H. Shalabi and G. Hadjisophocleous: "Case Study for proposed CANDU Fire Probabilistic Risk Assessment Model", 25th International Conference on Structural Mechanics in Reactor Technology (SMiRT 25) – 16th International Post Conference Seminar on "Fire Safety in Nuclear Power Plants and Installations" Ottawa, Ontario, Canada, October 27-30, 2019.
4) H. Shalabi and G. Hadjisophocleous: "Canada Deuterium Uranium Updated Fire Probabilistic Risk Assessment Model for Canadian Nuclear Plants", World Academy of Science, Engineering and Technology International Journal of Energy and Power Engineering Vol: 13, No: 10, 2019, Pages 480-484, 2020
5) H. Shalabi and G. Hadjisophocleous: "Canada Deuterium Uranium Updated Fire Probabilistic Risk Assessment Model for Canadian Nuclear Plants", ICFSST 2019: International Conference on Fire Safety Science and Technology, London, UK, September 25-26, 2019.
6) H. Shalabi and G. Hadjisophocleous: "HIGH Energy Arc Faults (HEAF) in Canadian Nuclear Plants", IFireSS 2019 – 3rd International Fire Safety Symposium, Ottawa, Ontario, Canada, June 5-7, 2019.
7) H. Shalabi and G. Hadjisophocleous: "CANDU Fire Load Densities in Canadian Nuclear Plants", Pages 171-177, CNL Nuclear Review, 2019
8) H. Shalabi and G. Hadjisophocleous: "CANDU Fire Database", Pages 179-189, CNL Nuclear Review, 18 December 2018.
9) H. Shalabi and G. Hadjisophocleous: "CANDU Fire Probabilistic Risk Assessment (PRA) Model", CNL Nuclear Review, Pages 97-108, 18 December 2018.

List of Acronyms

AEC - Atomic Energy Commission
AECL - Atomic Energy Canada Limited
AHJ - Authorities having jurisdiction
ANS - American Nuclear Society
ASME - American Society of Mechanical Engineers
BDBAs - Beyond design basis accidents
BWR - Boiling Water Reactor
C - Consequences
CANDU - Canada Deuterium Uranium
CCF - Common-cause failure(s)
CDF - Core damage frequency
CNSC - Canadian Nuclear Safety Commission
CSA - Canadian Standards Association
DC - Direct current
DOE - Department of Energy
ECI - Emergency coolant injection system
ECCS - Emergency core cooling system
EPRI - Electric Power Research Institute
EWS - Emergency water system
ET - Event tree
F - Initiating Fire Event Likelihood
FAQ - Frequently Asked Questions
FCDF - Fire induced core damage frequency
FDF - Fuel Damage Frequency
FDS - Fire Dynamics Simulator
FSSA - Fire safe shutdown analysis
FT - Fault tree
FP - Full power
FPS - Performance success probability
GCR- Gas Cooled Reactor
GDC - General Design Criteria
HEAF - High-Energy Arcing Fault
HEPs - Human error probabilities
HFEs - Human failure events
HRA - Human Reliability Analysis
HRRs - Heat release rates
IAEA - International Atomic Energy Agency
IFPRA - Implementation of Internal Fire Probabilistic Risk Assessment in Japan
ISRN - French Nuclear Safety and Radioprotection Institute
KKL - Leibstadt Nuclear Power Station
LOCA Loss of Coolant Accident
LOFA - Loss of Flow Accident
LP - Low power
LWRs - Light-Water Reactors

MCB - Main Control Board
MFW - Main feed water
MTTF - Mean time of failure
NEA - Nuclear Energy Agency
NEI - Nuclear Energy Institute
NFPA - National Fire Protection Association
NPP - Nuclear Power Plants
NRC - Nuclear Regulatory Commission
NRR - Nuclear Reactor Regulation
OECD - Organization for Economic Co-operation and Development
PNS - Probability of non-suppression
PRA - Probabilistic Risk Assessment
PSA - Probabilistic safety assessment
PWR - Pressurized Water Reactor
RES - Nuclear Regulatory Research
SCS - Shutdown cooling system
SD - Shutdown
SF - Severity factor
SG - Steam generator
SSC - Systems, structures and components
SSD - Safe shutdown
UCN 5&6 - Ulchin Nuclear Power Units
V&V - Verification and Validation
ZOI - Zone of influence

Chapter 1: Introduction

1.1 Problem Statement

Fire risk assessments results have indicated that fires are the major contributor to the risk of nuclear power plants. Therefore, fire risk assessment has become an integral part of the probabilistic safety assessment of nuclear power plants in addition to deterministic analyses [1].

The Electric Power Research Institute (EPRI) and the U.S. Nuclear Regulatory Commission (NRC) Office for Nuclear Research developed the NUREG/CR-6850 (Fire Probability Risk Assessment Methodology for Nuclear Power Facilities) methodology [2]. NUREG/CR-6850 is a state-of-the-art methodology to conduct fire Probabilistic Risk Assessments (PRAs) for commercial nuclear power plant (NPP) applications. NPPs in Canada and other countries (i.e.: Belgium, Czech Republic, Finland, Japan, Korea, and the Netherlands) use NUREG/CR-6850 or some portion of it in performing Fire PRAs. Other fire safety related NUREG guidelines are listed in APPENDIX A.

To properly use NUREG/CR-6850 for Canadian CANDU reactors, research is required to:

1. Assess the applicability of NUREG/CR-6850 to Canadian CANDU reactors, as the generic fire ignition frequencies provided in NUREG/CR-6850 were originally determined for the U.S. nuclear industry and thus may not represent those in other countries. There are differences in systems, structures, and components between CANDU reactors and U.S. reactors; e.g., some fires that may be negligible in light U.S. water reactors and are screened out by NUREG/CR-6850 may have consequences that are more significant in CANDU

reactors and vice versa, hence there is a need to define a Fire PRA methodology for CANDU reactors.

2. Assess the qualitative screening step in the Fire PRA that screens out fire scenarios that do not have merit for further quantitative assessment. This is a vital step in any Fire PRA. Different criteria are used in different parts of the world to identify the requirements for qualitative screening in a Fire PRA, and NUREG/CR-6850 uses the following criteria:

Components or cables: Fire zones may be screened out if they contain no components or cables associated with fire-induced initiating events and if they cannot lead to a plant trip due to plant procedures, automatic trip signal, or technical specification requirements. The justification for this criterion is not well documented, and this criterion does not take into account varying fire compartment configurations. Fire areas defined in a regulatory framework should fulfil the criteria of fire compartments in the framework of the Fire PRA. Fire zones may be defined in the framework of a fixed fire protection system, i.e., the zone of coverage. A fire zone may not fulfill the fire compartment definition [2].

Qualitative screening is performed to exclude fire scenarios from additional quantitative analysis based on the location of safety-related systems and equipment.

The issue in using the qualitative screening criteria listed above is:

- Screening out fire zones if they do not contain components or cables associated with fire-induced initiating events or if they do not lead to a plant trip due to either plant procedures, automatic trip signal, or technical specification requirements, is a very broad criterion and hard to fulfill accurately.

The criteria above do not consider varying fire zone configurations or ventilation conditions, physical and chemical properties of the fire load, and fire zone characteristics. The criteria above is not robust enough to account for other factors (e.g., fire spreading from an adjacent room or building, location of the target (initial fire location), fire detection and firefighting capabilities, etc.) in the screening step.

1.2 Fire Safety Goals in Canada

The Canadian Nuclear Safety Commission (CNSC) regulates the use of nuclear energy and materials to protect the health, safety, and security of Canadians and the environment, and to enforce Canada's international commitments on the peaceful use of nuclear energy. The CNSC regulates all nuclear-related facilities and activities including:

- Nineteen operating nuclear reactors at four sites
- Five uranium mines in Saskatchewan
- Eight processing and fuel fabrication facilities
- Major research facilities (governmental and university)

CNSC Fire Protection Goals:

- Health and safety of persons
- Protection of the environment and nuclear safety
- Achieving and maintaining a reactor in subcritical conditions
- Achieving and maintaining decay heat removal
- Maintaining the integrity of the fission product boundaries
- Limiting the release of radioactive materials that are located outside the reactor

CNSC Fire Protection Objectives:

1. Prevent fires from starting
2. Rapidly detect, control, and extinguish fires that do occur
3. Minimize the consequences of fires. Providing protection for systems, structures, and components (SSCs) is important for safety so that a prolonged fire will not prevent the safe shutdown of the plant

4. Control undesirable conditions in the nuclear facility and mitigate the consequences of severe accidents
5. Mitigate the consequences of significant releases of radioactive substances

Fire Safety Improvements Implemented at Canadian Nuclear Power Plants:

- CANDU NPPs were designed using the state-of-the-art codes, standards, and best international practices existing at the time of construction.
- Improvements in fire protection that assist Canadian NPPs to meet modern codes, standards, and best industry practices include:
    - Updated fire safety assessment in accordance with applicable standards.
    - Enhancements to operational practices that lead to measurable reductions in fire risk.
    - Complete safety upgrades, including design modifications such as diking around pumps, additional fire detection and suppression systems, fire barriers, shielding, etc.

1.3 Thesis Scope and Objectives

The scope of this thesis is to evaluate the use of NUREG/CR-6850 to perform Fire Risk Assessments (PRA) for Canadian CANDU reactors and to develop a CANDU Fire PRA. The Fire PRA is limited to demonstrate that the fire safe shutdown nuclear safety objectives and performance criteria are met. Therefore, the requirement to perform the fire safe shutdown functions is limited to plant areas where fires may have a potential impact on systems, structures, and components (SSCs).

The objectives of this thesis are to:

1. Analyze the CANDU Fire Database to explain the peak in fire numbers in certain periods and use this analysis to forecast future fires and predict initiating event's locations and causes and, evaluate the success probability of different fire protection systems.
2. Develop a Fire PRA methodology for CANDU reactors.
3. Create a fuel survey for all FSSA (fire safe shutdown analysis) rooms in CANDU reactors in Canada.
4. Investigate and analyze the fire safety threat due to High Energy Arcing Faults (HEAF) in CANDU reactors and make recommendations to mitigate this threat.
5. Develop a qualitative screening (decision tree) for CANDU reactors in Canada.
6. Demonstrate the use of CANDU Fire PRA methodology on two existing FSSA rooms in Canada.

1.4 Methodology

1. The analysis of the CANDU Fire Database involves investigating and analyzing Canadian CANDU fire events in the International Fire Data Exchange Project (OECD FIRE) of the OECD Nuclear Energy Agency (UNEA).
2. The main steps for developing the Fire PRA for the CANDU reactor are:
   a. Studying the general approach for Fire PRAs and NUREG/CR-6850
   b. Performing a full evaluation of NUREG/CR-6850 to assess its applicability to CANDU reactors. This evaluation will include an assessment of the adequacy of using the fire safe shutdown analysis (FSSA) list of credited SSCs for use in a Fire PRA when prepared in accordance with the requirements of NUREG/CR-6850.
   c. Studying international experience (e.g., from France, Finland, Germany, Japan, Korea, and Switzerland) experiences in applying NUREG/CR-6850, and the lessons learned
   d. Developing a Fire PRA methodology for CANDU reactors
3. The qualitative screening identifies fire zones that can be screened out from further assessments. The qualitative screening will be a decision tree that will either screen in fire scenarios for further quantitative analysis or screen out fire zones from further analysis. Creating a decision tree for CANDU involves the following:
   a. Conducting surveys to determine the fire load densities in about 1,230 FSSA fire zones at CANDU facilities where fires may have a potential impact on SSCs required to perform fire safe shutdown functions.

b. Conducting surveys to determine the ignition sources in zones where fires may have a potential impact on SSCs required to perform fire safe shutdown functions.

c. Conducting surveys to ascertain the fire protection equipment installed in zones where fires may have a potential impact on SSCs required to perform fire safe shutdown functions.

d. Categorizing all the fire zones into typical fire zones that resemble the surveyed CANDU plants' fire zones.

e. Using FDS to simulate fires in the typical fire zones to determine the impact of a fire on SSCs required to perform fire safe shutdown functions.

f. Creating a qualitative screening decision tree for CANDU reactors in Canada based on the simulation results.

4. Reviewing HEAF literature and providing recommendations to mitigate the associated risks.

5. Demonstrating the CANDU Fire PRA by applying it to existing two FSSA rooms.

1.6 Contributions to the Literature

This research contributes to the fire safety of CANDU reactors. There has been no similar work done on this topic previously. This research is divided into five parts. The first part is the development of the CANDU fire model to predict the FSSA cable damage probabilities in CANDU reactors in Canada. The CANDU Fire PRA results will contribute to design modifications in plants to enhance safety and thereby reduce its contribution to core damage frequency. The CANDU Fire PRA will improve the understating, prevention, and mitigation of the risks associated with nuclear operations and activities in Canada.

The second part of this research is the CANDU fire database. The author contributed to the development and the collection of the fire database as a part of the Engineering Design Analysis team at CNSC. The author analyzed the spikes and trends in the fire database and predicted future fires related to CANDU nuclear reactors in Canada. The data collected and analyzed will contribute in identifying CANDU plants´ vulnerability related to fire safety, predicting future fires and finding fire-related trends.

The third part of this research is the extensive fuel survey that was carried out on 1,230 CANDU FSSA fire zones for all five operating nuclear sites in Canada. The survey included the floor areas/ceilings heights, ceiling/wall/floor construction, the minimum and maximum fuel loads, minimum and maximum fuel load densities, percentages of fire zones that have fire suppression, fire detection, access to fire hoses and portable fire extinguishers. The survey also included fire compartment that have mechanical ventilation systems, types of ignition sources, types and quantities of combustibles and identified fire zones with potential HEAF. This is a very important piece of work as it

compares all different fire zones in the Canadian nuclear plants per function and will be used as a reference document to evaluate similar fire zones in the future. This fuel survey was also used in the next part of this research to create a new qualitative screening step in the CANDU Fire PRA.

The fourth part of this research is creating a new qualitative screening step in the CANDU Fire PRA. After analyzing both the fire database and the fuel survey, FDS simulations were carried out to assess the impact of selected fire scenarios on FSSA cables. The FDS results were used to construct a decision tree and to plot (combustible loads vs. areas) curves. The plotting of these (combustible loads vs. areas) curves resulted in the identification of different area thresholds for corresponding combustible loads. This is a very important part of this thesis as it relates thresholds for combustibles to areas. All CANDU operators can use this decision tree and save a lot of time, effort and money that was previously used in assessing all FSSA fire zones that are screened out by the decision tree. In addition, it will also assist CANDU operators in assessing new fire zones and can be used as a design reference document for fire safety.

The last part of this research identified HEAF as a new fire-related threat to CANDU plants, and analysed international and national HEAF fire events. The research also discussed gaps in HEAF prediction models in NUREG/CR-6850 and the international OECD-NEA projects (Database Project FIRE and experimental Project HEAF) efforts to develop an accurate prediction model for HEAF. The research includes recommendations to mitigate the occurrence and the consequences of HEAF fire events in Canada and internationally. CANDU operators can adopt the recommendations to mitigate potential HEAF fire events until a HEAF correlation equation is developed.

Chapter 2: Literature Review

In this chapter, the results of the literature review are described in order to summarize the literature related to NUREG/CR-6850, fire load survey methods, fire modeling, the applicability of NUREG/CR-6850 to Canadian Reactors, qualitative screening approaches and NUREG/CR-6850 CANDU Application Gaps.

2.1 NUREG/CR-6850 EPRI/NRC-RES Fire PRA Methodology

In 2004, the U.S. Nuclear Regulatory Commission (NRC) revised the regulations to allow licensees to voluntarily shift to risk-informed fire protection under NFPA 805 [3]. There are 47 reactors in the U.S. (out of 104 reactors operating in the U.S.) which were planning to shift (or were in the process of shifting) to a more risk-informed and performance-based approach [3]. The use of PRAs in the nuclear safety arena contributes to (i) decision-making that is enhanced by the use of PRA insights; (ii) more efficient use of resources; and (iii) reduction in unnecessary burdens on licensees. In order to benefit from these advantages, PRAs should be used "to reduce unnecessary conservatism" and the PRA output "should be as realistic as possible" [2].

The Electric Power Research Institute (EPRI) and the NRC developed NUREG/CR-6850 [2], which is a state-of-the-art methodology for Fire PRAs for NPP applications. NPPs in Canada and other countries (e.g., France, Finland, Japan, Korea, and Switzerland) use NUREG/CR-6850 or some portion of it in performing Fire PRAs.

The core damage frequency (CDF) is the direct frequency of the damage to the reactor core according to the sequence in the risk model. A fuel damage frequency (FDF) can be defined as the fuel loss of integrity on site, which can cause a severe accident,

regardless of the plant operational state of the reactor or location of the fuel. This thesis is limited to CDF.

Overall, NUREG/CR-6850 involves 16 interconnected tasks [2]. The comprehensiveness of the resulting PRA makes it an efficient tool for identifying hazards and assessing risks as well as for establishing controls in nuclear plants. A description of each task is provided below:

Task 1: Plant Boundary and Partitioning

The goal of this task is to identify both the "global plant analysis boundary", which encompasses all areas on the site with the potential to contribute visibly to fire risk, and the "fire zones" that will serve as the basic physical analysis units for the Probabilistic Safety Assessment (PSA).

Task 2: Fire PRA Component Selection

This task mainly identifies the initiators, sequences, components, and failure modes to be modeled by the PSA. The analysis makes use of both the fire targets and fire ignition sources. Component selection begins with an evaluation of the internal events. A series of additional reviews follows this evaluation (e.g., reconciliation with fire-safe shutdown analysis, multiple spurious operations reviews, etc.), intended to identify fire risk-relevant failures that may not have been captured by the internal events model.

Task 3: Fire PRA Cable Selection

This task identifies the circuits/cables associated with the components identified as targets in Task 2 and their corresponding routing through the plant whose fire-induced failure could affect equipment on the fire PSA component list.

Task 4: Qualitative Screening

This task screens out fire compartments from additional consideration based on the following criteria:

- Fires originating within the fire zone will not cause a reactor trip (either automatic, manual, or forced shutdown)
- Fires originating within the fire zone will not affect Fire PSA components or cables

Task 5: Fire-Induced Risk Model

The full scope of this task, as stated in NUREG/CR-6850, is to assess the internal events in the PRA model to quantify fire-induced risk. This task develops a model and quantification process for determining fire-induced core damage frequencies (CDF).

Task 6: Fire Ignition Frequencies

This task first identifies every credible ignition source within the global plant analysis boundary, and afterwards, the fire frequencies for each ignition source and each fire compartment are calculated.

Task 7: Quantitative Screening

The primary objective of this task is to provide the user with an approach to quantify the Fire PRA Model developed in Task 5 and to screen out fire zones based on quantitative screening criteria. This task uses quantitative criteria to screen low-risk fire zones from further consideration.

Task 8: Scoping Fire Modeling

Task 8 identifies and screens fixed ignition sources. Ignition sources are binned into categories and analyzed in order to determine a bounding zone of influence (ZOI). The fire

ZOI concept was originally introduced as part of the Electric Power Research Institute's (EPRI) FIVE methodology and has been carried forward into the methodology presented in NUREG/CR-6850 [2]. The ZOI is defined as the region where a given target is expected to be damaged by fire-generated conditions. This task allows the analyst to screen out the fraction of fire frequency that is not anticipated to cause damage beyond the ignition source itself.

Task 9: Detailed Circuit Failure Analysis

The purpose of this task is to determine component responses in each unscreened fire zone to postulate cable failure modes in order to screen out cables that do not affect component operability. In the cable selection task, all cables associated with a particular component may be assumed to fail basic events associated with that component. However, this assumption may be determined during the quantification of the initial fire PSA as too conservative for specific components. For those components, this task performs a detailed review of circuit and cable documentation to identify the specific cables that will cause specific failures of concern.

Task 10: Circuit Failure Mode and Likelihood Analysis

This task develops and applies a conditional probability that, if a cable is damaged by fire, a hot short will occur on any specific cable of concern. It is applied primarily to fires related to a significant risk of fire-induced spurious operations.

Task 11: Detailed Fire Modeling

Detailed fire modeling uses computer simulations to produce estimates of target set mean times of failure (MTTFs) and target set failure probabilities. This assignment involves three tasks: 1) Single fire analysis; 2) Multi-fire analysis; 3) Main Control Room

(MCR) analysis. Detailed fire modeling determines the probability that a target will fail given a specific initiator. This process generates a severity factor as a function of the fire ignition source and the component, where the severity factor is the conditional probability of a component's failure along with a mean time to failure.

Task 12: Post-Fire Human Reliability Analysis (HRA)

The post-fire HRA assesses the impact of fire scenarios on the human actions addressed in the base PRA or random events PRA model that are used in the Fire PRA model. This task starts with identifying human failure events (HFEs) to include in the fire PSA. Then, human error probabilities (HEPs) are calculated based on the potential fire-induced impact on the performance-shaping factors (e.g., cues, stress, travel path, etc.) associated with each HFE. Finally, a recovery rule process is developed to apply each relevant HFE to the fire cut sets.

Task 13: Seismic-Fire Interactions

This task consists of qualitative evaluations of seismically induced fires, seismic degradation of fire suppression equipment, and seismic actuation of fire suppression systems.

Task 14: Fire Risk Quantification

This task quantifies the fire-induced core damage frequency (CDF) for each fire scenario developed in the previous steps. The Fire PRA[1] results, including significant fire risk contributors and fire risk insights, are identified and documented.

[1] Please note that PRA and PSA are used synonymously in this thesis.

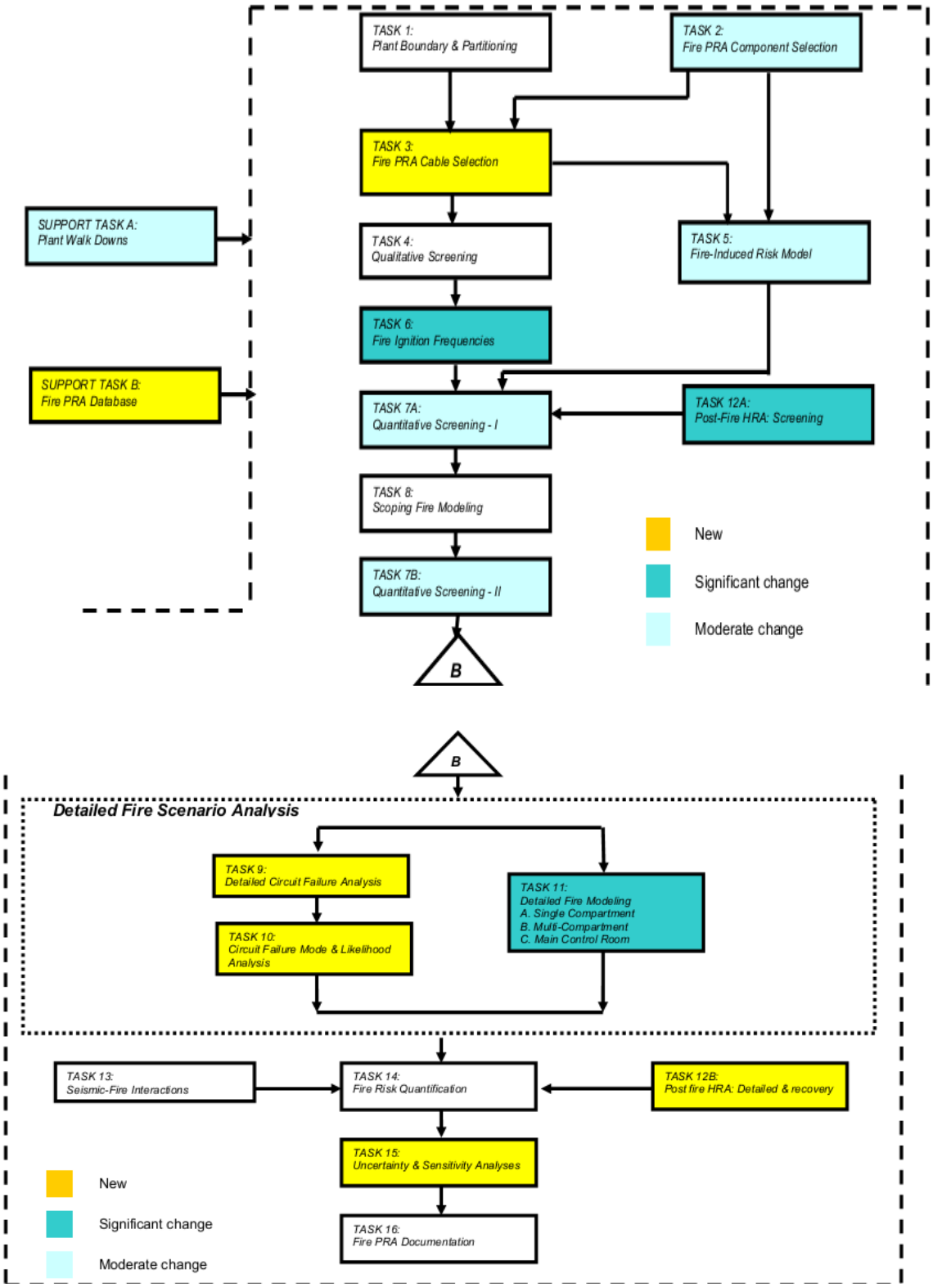
Task 15: Uncertainty and Sensitivity Analysis

This task involves a Fire PRA uncertainty analysis to the extent afforded by current technology. Calculating the uncertainty of the various basic events used within the model (ignition frequencies, severity factors, non-suppression probabilities, and human factors) is handled outside of the model.

Task 16: Fire PRA Documentation

This task involves documenting the Fire PRA and its results, in order to confirm that there is satisfactory documentation of the Fire PRA to permit a review of the Fire PRA's development and results, as well as to provide a written basis for any future uses of the Fire PRA. In Figure 1, the yellow boxes indicate a new fire scenario with all other associated new data inputs (refer to all yellow boxes). The light blue boxes reflect the moderate changes associated with every input of new fire scenario analysis. The dark blue boxes reflect the significant changes associated with the data input.

Figure 1. Fire PRA NRC Regulatory Guide 6850 Task 1-16 Breakdown Diagram [



Important changes have occurred in Fire PRA over the years. Methods, models and computational tools for performing Fire PRA have improved, more fire-related operational experience has been recoded, and experimental results have been generated [4]. An important change in the modern fire PSA is the consideration of combination of events. The modern Fire PRA evaluates different event combinations or casualty related events, for example: flooding-fire events, HEAF-fire events, explosion-fire events and other events-fire events, rather than only assessing seismic-fire interaction in task 13 of NUREG/CR-6850 [2]. HEAF fire event[2] is high energy, energetic or explosive electrical arc fault. NUREG/CR-6850 provides a general process to conduct Fire PRAs for NPPs. There are three main steps for NUREG/CR-6850: i) fire ignition frequency estimation associated with; ii) fire scenario occurrence frequency, where detailed fire modeling is involved and the failure probability of fire protection equipment is considered; and iii) fire risk quantification of the Fire PRA model to produce the fire risk results. Fire risk quantification should be backed up by a fire database, including ignition frequencies, and fire protection equipment.

[2] High Energy Arc Faults (HEAF) are energetic or explosive electrical equipment faults characterized by a rapid release of energy in the form of heat, light, vaporized metal and pressure increase due to high current arcs between energized electrical conductors or between energized electrical components and neutral or ground. HEAF events may also result in projectiles being ejected from the electrical component or cabinet of origin and result in fires.

2.2 Fire Load Density and Fuel Load Surveys

Engineering calculations and tools, including both computer models and experiments, are used to show that performance-based designs are acceptable by demonstrating that the designs meet the established performance criteria. The design fires used in the fire scenarios selected for the calculations are an important parameter and shall represent fires that may occur in the various spaces considered. In addition to design fires, performance-based designs must consider how a building's structure and fire protection systems would perform in the event of a fire. The design fire depends on the total amount of fuel and its arrangement and distribution in the space as well as the geometry and ventilation characteristics of the fire zone in which the fire originated, all of which govern the intensity and duration of the fire [5-7].

The quantity of combustibles in a fire compartment is commonly expressed as the total heat energy (MJ) that can be released through fully ventilated combustion and is known as the fire load. The fire load density is usually expressed as the total fire load per unit of internal surface area (MJ/m^2) or floor area, and largely depends on the occupancy. It is suggested that both fixed and transient fire loads should be taken into consideration [8] when calculating the fire load.

Survey Methods

NFPA Standard 557 [9] states that either the inventory method or the weighing method or a combination of both can be used to conduct a fire load survey. The weighing method requires physical entry into a building by a surveyor to document the contents and characteristics of all items within a fire zone, while the inventory method is able to

account for both fixed and content fire loads in a fire zone. The combination method has been found to produce the most accurate values for fire loads [5].

Inventory method. The inventory method involves the calculation of the mass of a combustible by using its measured volume and corresponding density. This method requires the physical entry of a building or fire zone to quantify all combustible items within it. The combustible energy is then calculated by using the net heat of combustion of the fuel package [10, 11] determined by the collection of visual data. This concept assumes that there is a relationship between the visual characteristics of the combustibles and their weight. This relationship is called a “transfer function” or “formula for weight”, which is expressed in terms of physical characteristics, and masses of combustibles are expressed based on their transfer functions.

Weighing method. The weighing method also requires physical entry into a building or fire zone to detail the contents and characteristics of the combustibles inside, but different from the inventory method, the masses of the combustibles are obtained by direct weighing. The use of the direct weighing method may not be as common as the inventory method but is often used in conjunction with the inventory method.

Combination method. The combination method combines the use of the direct weighing method and the inventory method. This can include an inventory of pre-weighed combustibles and the calculation of mass based on direct measurement of volume and material densities. Using the inventory or direct weighing method alone has several disadvantages that may obstruct the progress of the survey and negatively affect the survey results. This has resulted in the use of both methods for a number of surveys in the past [8, 12-15].

This research will follow the NFPA 557 combination method in carrying out the fuel load surveys for CANDU nuclear plants. The total fire load in a fire zone is calculated using the following equation:

$Q = \Sigma\, k_i\, m_i\, h_{ci}$ **(Equation 1)**

Where Q = total fire load in a fire zone (MJ)

k_i = proportion of content or building component i that can burn

m_i = mass of item i (kg)

h_{ci} = calorific value of item i (MJ/kg)

The total fire load per unit area in a fire zone is calculated using the following equation:

$Q'' = Q / A$ **(Equation 2)**

Where Q" = Fire load per unit area (MJ/m^2)

A = floor surface area of a fire zone (m^2)

The survey includes multiple similar fire zones, and hence the average and standard deviations of the fire load are computed. The standard deviation is calculated using the following equation:

$\sigma = (\Sigma\, (x_i - x)^2 / N)^{1/2}$ **(Equation 3)**

Where σ = standard deviation

x_i = fire load from i^{th} sample

$\overline{x}$ = average of all fire load samples

N = number of fire load samples

The best practice nuclear fuel survey method was followed in all 1,230 fire zones. NFPA 557-combination method mean values [9] were introduced for binning the fire zones per

function, in an effort to figure out the fire simulations criteria to develop the qualitative analysis step in the Fire PRA.

2.3 Fire Modeling

Fires are a major contributor to nuclear power plant risks, and the main risk in the fire PSAs is fire-induced damage to control and power cables. Fire protection relies on fire modeling to define the consequences of fires [16]. Verification and validation have been performed on five selected models used in the nuclear industry: (1) the NRC's NUREG-1805 Fire Dynamics Tools (FDTs); (2) EPRI's Fire Induced Vulnerability Evaluation (FIVE); (3) National Institute of Standards and Technology's (NIST's) Consolidated Model of Fire Growth and Smoke Transport (CFAST); (4) Electricité de France's (EdF's) MAGIC; and (5) NIST's Fire Dynamics Simulator (FDS) (as shown in the U.S. NRC Verification and Validation study [17]).

Usually, validation includes comparing modeled results with experimental measurements. Differences that cannot be justified in terms of numerical errors and/or uncertainty in the measurements in the model are acknowledged as being due to the simplicity of the physical models, while model verification involves testing for mathematical robustness and accuracy.

- FDTs are a group of empirical correlations in spreadsheet form. Mostly, the correlations in an FDT library are closed-form algebraic expressions coded in spreadsheets to deliver a user-friendly interface that decreases input and computational errors.
- FIVE is an alternative library of engineering calculations in spreadsheet [18]. FIVE contains functions coded in Visual Basic.

- CFAST is a two-zone fire model that predicts the environment that arises within fire zones for a fire described by the user. CFAST was developed and is maintained by NIST's Fire Research Division [19].
- MAGIC is a two-zone fire model developed and maintained by EdF. It is available to its members through EPRI [20].
- FDS is a Computational fluid dynamics (CFD) model that was developed and maintained by the National Institute of Standards and Technology (NIST) [21].

Overall, computational fluid dynamics (CFD) models such as FDS can provide more accurate and detailed results than empirical and zone models. The zone models CFAST and MAGIC predict global quantities and fire compartment pressures well, but they are less accurate than FDS in predicting localized quantities, such as ceiling jets, surface temperatures, and heat flux [17].

CFD models (also called field models) are one of the most sophisticated tools available to fire safety engineers. These models are based on fundamental laws of physics rather than empirical correlations. As the use of fire modeling increases in support of day-to-day NPP applications and fire risk analyses, the importance of verification and validation (V&V) also increases. The use of V&V builds confidence in a model by evaluating its underlying assumptions, capabilities, and limitations, as well as by quantifying its performance in predicting fire conditions that have been measured in controlled experiments [21, 22].

FDS is a computer program that solves the governing fluid dynamics equations with a particular emphasis on fire and smoke transport, while Smokeview is a companion program that produces images and animations of the FDS calculations [23]. FDS is a

CFD model for fire-driven fluid flows that numerically solves a form of Navier-Stokes equations appropriate for low-speed, thermally driven flows with an emphasis on smoke and heat transport from fires. The partial derivatives of the conservation equations for mass, momentum, and energy are approximated as finite differences, and the solution is updated in time on a three-dimensional, rectilinear grid [23, 24].

In CFD, the governing equations include Navier-Stokes equations and mass-, energy-, and species-conservation equations, which can be applied to both laminar and turbulent flows. The area of interest is divided into many small volumes, and the equations are discretized into algebraic equations. Furthermore, the turbulence of the fluid flow must also be solved, and using turbulence models typically does this, because direct solving of the turbulent structure demands very large computational times.

FDS can reliably predict gas temperatures, major gas species concentrations, and zone pressures to within about 15 %. It can also consistently predict heat fluxes and surface temperatures to within about 25 % [22]. FDS requires as inputs the geometry of the building compartment being modeled; computational cell size; location of the ignition source; ignition source; thermal properties of the walls; furnishings and their composition and size; location; and openings to the outside, which critically influence fire growth and spread [23].

FDS will be used in this research for its ability to analyze very detailed scenarios, which require high precision for implementing the fundamental fluid dynamics equations over complex domains while taking into consideration aspects such as turbulence descriptions, reaction kinetics, radiation transport, and pyrolysis. FDS also permits users to model interactions that happen simultaneously in a fire accident, helping to evaluate

the effect of different parameters on the event progression. It is possible to add sprinklers and smoke detectors to the simulation simply by describing their position and is the same as scribing their properties in the Cartesian coordinate system [22, 24].

2.4 Applicability of NUREG/CR-6850 to Canadian Reactors

Due to differences between Canadian and U.S. nuclear reactor types, the applicability of NUREG/CR-6850 to CANDU reactors is not clear. For example, the generic fire ignition frequencies provided in NUREG/CR-6850 reflect the experiences only of the U.S., and not Canada. There are also differences in systems, structures and components when comparing Canadian nuclear reactors – which use pressurized heavy water, to U.S. reactors – which use light water. The key difference between the two types is the moderator, which uses ordinary water (H_2O) in light water reactors and deuterium oxide (D_20) in heavy water reactors. The water moderator also functions as a primary coolant in light water reactors, while CANDU reactors use either light or heavy water coolant and the moderator and coolant are separate.

CANDU reactors have a number of inherent safety features that differentiate them from light water reactors. For example, the subdivision of the core reduces the reactivity effect of a loss-of-coolant accident (LOCA) and carries two core-passes per loop. The moderator provides a low-pressure environment for the control rods. Because heavy-water neutron kinetics are slower than light-water kinetics by several orders of magnitude, control is easier and power refuelling is possible [25].

NUREG/CR-6850 refers to combinations of fire locations and equipment types (ignition source and frequency) as bins. NUREG/CR-6850 provides the list of these bins and their respective generic mean frequencies in terms of the number of fire events per reactor year. Below are some examples of the NUREG/CR-6850 fire bins that are not applicable to CANDU:

- Bin 3, Transient and hot work within the containment: This bin is for the large dry type containment used in US pressurized water reactor (PWR) facilities. CANDU does not have a comparable "containment" structure, and therefore this bin is not utilized.
- Bin 20, Off-Gas hydrogen recombiners for boiling water reactors (BWRs): In general, this bin is applicable to hydrogen recombiners in BWR off-gas systems and is not applicable to the CANDU design. Although the Bruce Power Station does have 'recombination ' units, this equipment does not meet the intent of the NUREG/CR-6850 bin 20.
- Bin 22, RPS MG Sets: NUREG/CR-6850 specifies this equipment for pressurized water reactor-type plants. There are no reactor protection motor generator (RPS MG) sets in CANDU.

In addition, LWRs have other systems that do not exist in CANDU plants. Consequently, fire events that are related to these types of systems in LWR plants are not considered in CANDU plants. For example, fires in the following LWR systems are screened out:

1) Turbine-driven main feedwater pumps in boiling and pressurized water reactors (BWRs and PWRs). However, pump-only-related fires are retained (e.g., those caused by lube oil or overheated bearings).
2) Emergency feedwater pumps in PWRs
3) Turbine-driven high-pressure core injection system and turbine-driven reactor core isolation cooling system in BWRs
4) Standby gas treatment system in BWRs

In terms of regulating fire protection in nuclear plants, the fire protection rules of PWRs (R.G 1.189) [26] express a set of required systems and components for achieving a hot shutdown. This set of required systems and components must be free of fire damage when a fire breaks out in a nuclear power plant regardless of the concurrent failure of multiple systems and components required for a cold shutdown, and at least one system must be repaired and restored within 72 hours. In CAN/CSA-N293-12 [27], the fire protection rules for pressurized heavy water reactors do not have different rules associated with different systems and components for hot or cold shutdowns, and they state that the capability to perform safety functions must be protected in order to achieve a safe shutdown under all operation modes.

2.5 Qualitative Screening

The qualitative screening Task 4 in NUREG/CR-6850 screens out fire scenarios that do not merit further quantitative assessment. Different criteria are used in different parts of the world to identify requirements for the qualitative screening part in Fire PRAs. NUREG/CR-6850 uses the following criteria: Components or cables: Fire zones are screened out if they contain no components or cables associated with fire-induced initiating events and if they cannot lead to a plant trip due to plant procedures, an automatic trip signal, or technical specification requirements.

The screening process to identify critical fire zones is an important first step in a fire risk assessment. Qualitative screening analysis should not be too conservative so that a very large number of fire scenarios remains for the detailed quantitative analysis. Universal challenges are reflected in the qualitative tasks. These tasks include rationales for fire-fire zone partitioning, identification, and screening of the most potentially affected fire zones.

Reasonable screening values should be used when information has yet to be determined. As new information becomes available and detailed analyses are completed, the model can be refined and the screening values can be replaced by realistic values. It is easy to identify whether a postulated fire scenario will result in a fire-induced initiating event. However, cable data for all equipment in the plant is not typically identified, and therefore, for many scenarios, it is uncertain what might be the effect of a fire, if any at all. NUREG/CR-6850 guidance acknowledges that for these cases, a fire-induced initiating event may have to be assumed. Typically, at a minimum, a plant trip will be assumed [28].

The justification for this criterion is not well documented, and this criterion does not take into account varying fire zone configurations. Screening out fire zones if they do not contain components or cables associated with fire-induced initiating events and if they do not lead to a plant trip due to either plant procedures, an automatic trip signal, or technical specification requirements, is a very broad criterion and is challenging to fulfill.

2.6 NUREG/CR-6850 CANDU Application Gaps

The applicability of NUREG/CR-6850 to the Canadian CANDU reactors has not been assessed, as the generic fire ignition frequencies provided in NUREG/CR-6850 represent only the U.S. industry experience. As noted previously, there are differences in systems, structures, and components between CANDU reactors and U.S. reactors, and some fires that are negligible in U.S. LWRs and are screened out by NUREG/CR-6850 may have more significant consequences in CANDU reactors, and vice versa.

All fires that occur at CANDU reactors are reported to the OECD FIRE Database. The objectives of the OECD FIRE Database Project involve multilateral (Belgium, Canada, Czech Republic, Finland, France, Germany, Japan, Korea, Netherlands, Spain, Sweden, Switzerland, UK and the USA) co-operation in the collection and analysis of data relating to fire events.

Meanwhile, the OECD FIRE Database is in phase 6 of the project since January 2020. This thesis covers fire events up to the end of 2017. A total of 400 nuclear reactor units in twelve member countries (Canada, Czech Republic, Finland, France, Germany, Korea, Japan, Netherlands, Spain, Sweden, Switzerland, and USA), with more than 500 fire events have been reported already [29]. Data collection is continuously ongoing, with an approximate average of 30 expected events per year being included. This may increase significantly as soon as the several hundred fire event records from the U.S. FEDB can be submitted [29].

An evaluation of the applicability of NUREG/CR-6850 is required for CANDU reactors, including an assessment of the adequacy of using its fire safe shutdown analysis list of credited structures, systems, and components (when prepared in accordance with

the requirements of CSA N293-12) for use in the Fire PRA (when prepared in accordance with the requirements of NUREG/CR-6850). The evaluation must also include a review of NUREG/CR-6850's applicability to the Canadian Nuclear Safety and Control Act and the General Nuclear Safety and Control Regulations, CSA N293-12 "Fire Protection for CANDU Nuclear Power Plants."

Chapter 3: CANDU Fire Database

The OECD /NEA member countries: Canada, Czech Republic, Finland, France, Germany, Japan, Spain, Sweden, Switzerland, and USA decided already in the early 2000s to create the International Fire Incidents Records Exchange (OECD FIRE) Project to encourage multilateral co-operation in the collection and analysis of data related to fire events in nuclear power plants. The objectives of the Project are according to [29] and [30]:

- Record fire event characteristics to simplify quantification of fire frequencies and fire scenario frequencies,
- Collect and analyze fire events in order to better comprehend such events and their causes, and develop methods to prevent them from occurring in the future,
- Generate qualitative insights into the causes of fire events, which can then be used to develop methods or mechanisms for their prevention or for mitigating their consequences,
- Create a mechanism for the efficient feedback of fire-related experiences, including the development of defenses against their occurrence, such as indicators for risk-based inspections.

The scope of the OECD FIRE Database Project is limited to fires that occur at nuclear power plants. Fires at research reactors, nuclear waste storage facilities, etc. are not in the scope. The reporting includes all on-site internal fires (inside and outside buildings) as well as external fires if these have the potential to impact nuclear safety. The reporting includes fires in all operational modes and fires during construction and decommissioning.

The main aim of the OECD FIRE Database Project is to provide a platform for multiple countries to collaborate on fire event data and thereby enhance the knowledge of fire phenomena. In addition, it aims to improve the quality of fire-related risk assessments. For applications with Fire PRAs, the OECD FIRE Database is also able to provide component-specific fire frequencies for different reactor types and plant modes of operation including low power and shutdown scenarios, and for the decommissioning phase, if data are reported during decommissioning. Several activities have been conducted to apply this Database for analysis of the operating experience with fires in nuclear power plants reported to the OECD FIRE Database. Examples include the analysis of HEAF induced fires, combinations of fires with other anticipated events and hazards [31], and the analysis of apparent and root causes of fires in nuclear power plants [32].

The events reported in the OECD FIRE Database are from the early 1980s up to the end of 2019. This database provides a good platform for starting the analytical phase, and it is now possible to quickly estimate fire frequencies for different types of fire events for all plant operational states and different types of reactors. Canada reported 76 fires between the years 1981 and 2017. There were two fires in Canadian NPPs from 2013 to the end of 2017 representing the end of the reporting period in Canada for the 2017 database version [29], including an event from 2012 has meanwhile been included in the Database. There was another fire (2018) between 2017 and the end of 2019 in Canada, which were already reported and will be part of the next database version.

The data from nuclear installation operation experiences show combinations of fire and other events seen during the lifetime of these installations. The necessary

functions of the systems, structures, and components (SSCs), which are of huge importance for safety, could be impaired if such combinations occur due to degradation or loss of their primary functions. This is the reason why it was decided to explore combinations of fires and other hazards. The investigation requires differentiating between three types of combinations: fire and consequential event, event and consequential fire and fire and independent event occurring simultaneously.

Figure 2 shows the number of nuclear plant fires reported per country, and Figure 3 shows the number of fires by reactor type for 1981 -2017[3]. Figures 4-12 show graphs for each variable, summarizing the frequency by percentage for the three nuclear reactor types: Boiling Water Reactor (BWR), Pressurized Heavy Water Reactor (PHWR), Gas Cooled Reactor (GCR) and Pressurized Light Water Reactor (PWR) [29].

[3] Not all countries reported between 1981 to 2017. The number of observed reactor years will be provided later on to illustrate more comparable comparison between operating experiences

Figure 2. Fires reported per country from the FIRE Database as reported in [29]

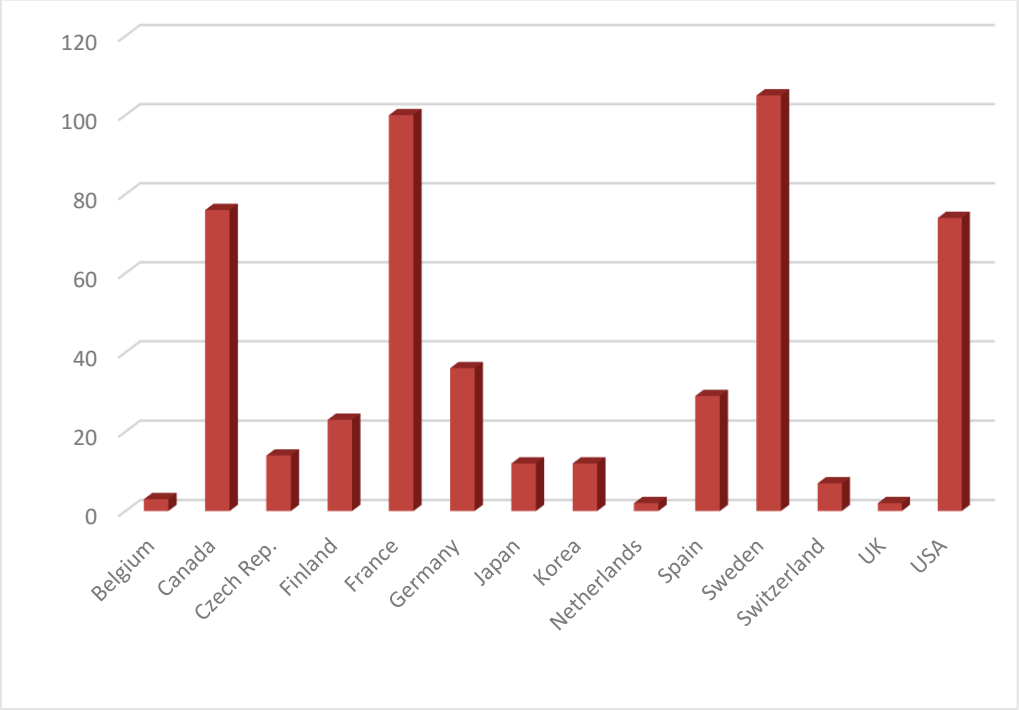


Figure 2 shows 76 fires for Canada compared to Korea (12 fires), USA (74 fires), Switzerland (7 fires), Sweden (105 fires), Spain (29 fires), Netherland (2 fires), Japan (12 fires), Germany (36 fires), France (100 fires), Finland (23 fires), Belgium (3 fires), UK (2 fires) and Czech Republic (14 fires).

Remark: Belgium and the UK joined the OECD FIRE Database project in phase 5, and hence only few fires from recent years had been reported. The experience from these countries cannot be applied for a direct comparison or statistical use.

Figure 3 shows that PHWRs had a total of 66 fires for in total 597.08 reactor years of observation, compared to PWRs with 258 fires over 5530.82 years of observation , GCRs with 5 fires for an observation period of 127.66 years, and BWRs with 154 fire over 2785.99 years of observation. While Canada has 76 + 1 reported fires representative for 552.14 reactor years, 64 + 1 of these fires were reported for CANDU

reactor fires representing an observation period of 547.90 reactor years, and 12 fires were reported from the Gentilly 1 site (representing 4.24 years of observation), which had prototype CANDU-BWR reactors, based on the SGHWR design.

Figure 3. Fires by reactor type up to 2017 [29]

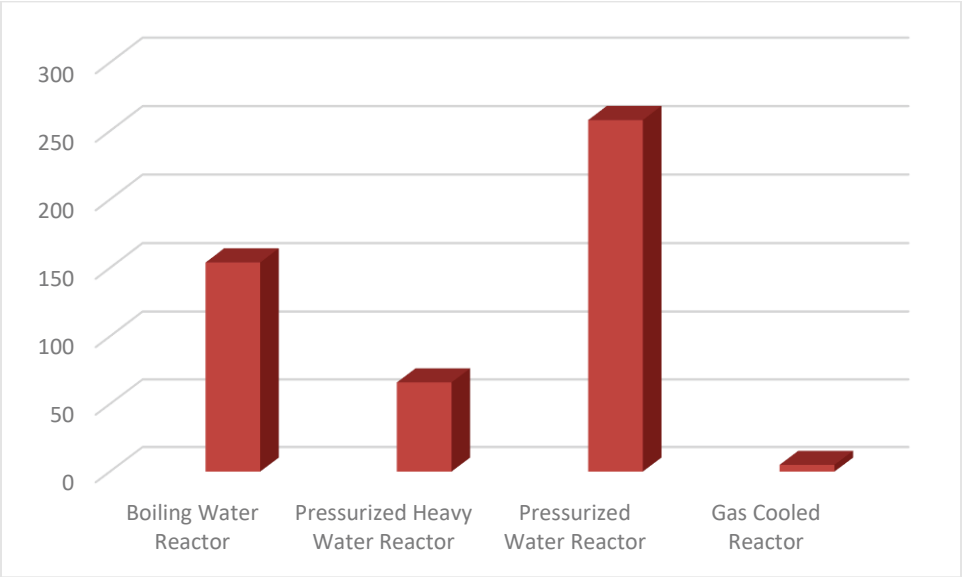


Table 1. Operation mode prior to fire [29]

Operation mode	BWR	GCR	PHWR	PWR
Construction phase	1	0	0	3
Decommissioning	0	0	0	0
Hot stand-by	0	0	1	7
Long term safe shutdown	0	0	0	0
Power operation	102	3	36	170
Shutdown mode	39	1	8	64
Start-up mode	8	0	1	13
Unknown	4	1	29	1

Table 1 and Figure 4 show the operation mode of the four types of nuclear reactors prior to a fire. As it can be seen in the table, the highest percentage for the operation modes for all three types of reactors prior to fire is “Power Operation”. While the lowest operation mode for all types of reactors prior to fire is during the “Decommissioning” mode. For decommissioning mode, there are meanwhile events, but only few countries report fire during decommissioning.

Figure 4. Operation mode prior to fire [29]

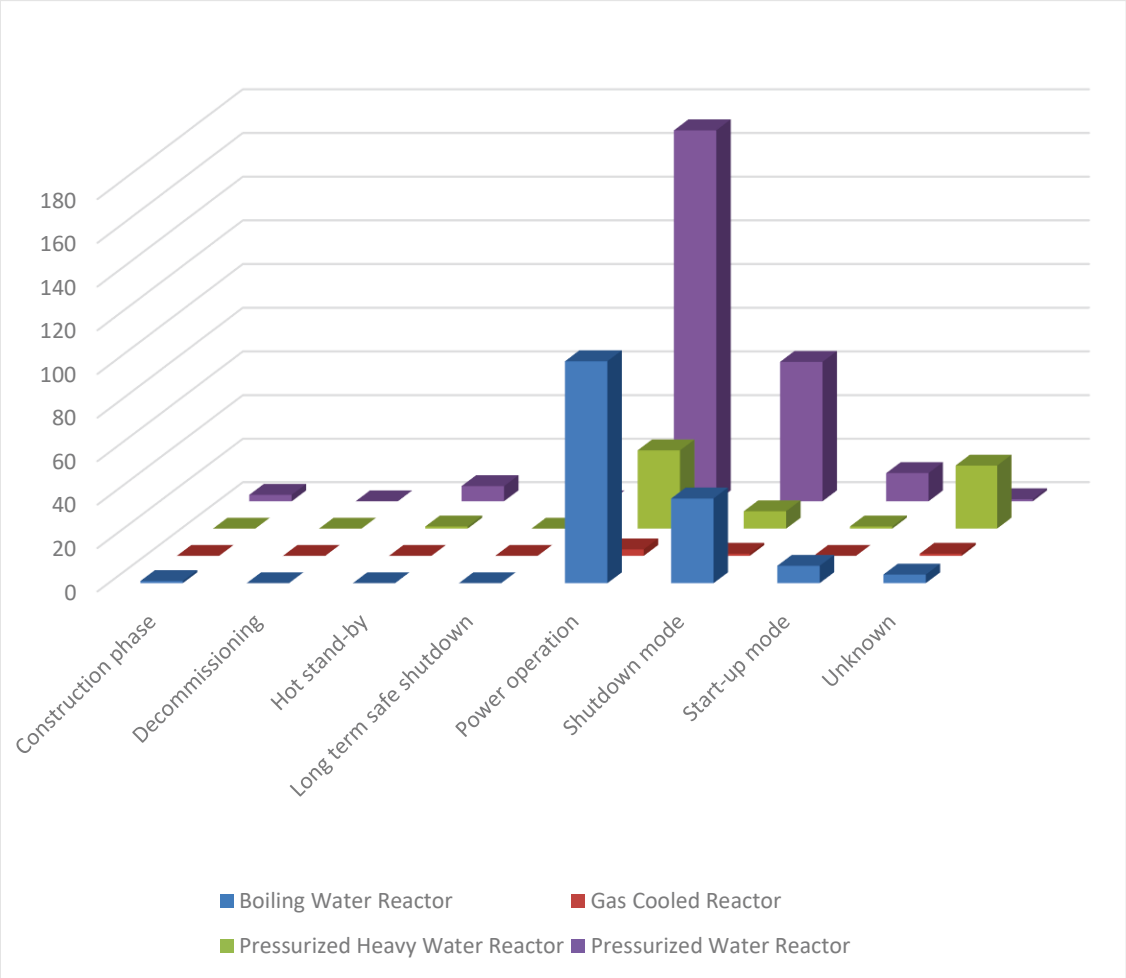


Table 2. Operational mode after ther fire [29]

Operational mode	BWR	GCR	PHWR	PWR
Construction phase	1	0	0	3
Decommissioning	0	0	0	0
Hot stand-by	12	0	1	42
Long-term safe shutdown	0	0	0	0
Power operation	59	2	28	95
Shutdown mode	72	2	17	110
Start-up mode	-	0	0	7
Unknown	6	1	29	1

Figure 5. Operational mode after fire occurrence [29]

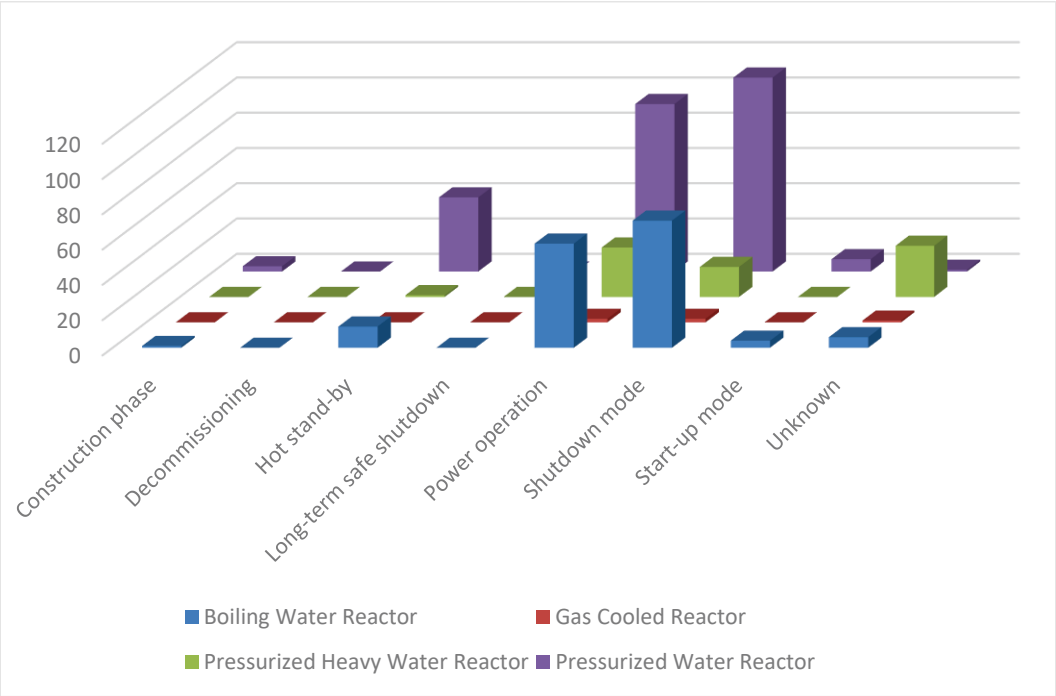


Table 2 and Figure 5 show the nuclear reactor's operational mode as consequence of the fire events for all four types of nuclear reactors. The operation mode "Shut Mode" has the most events for BWR, GCR and PWR during fire initiation compared to "Power Operation" for PHWR. While the lowest operational mode during fire initiation for all three types of reactors is the "Decommissioning". There are only few countries that report fires during long-term safe shutdown.

Figure 6. Building where fire started [29]

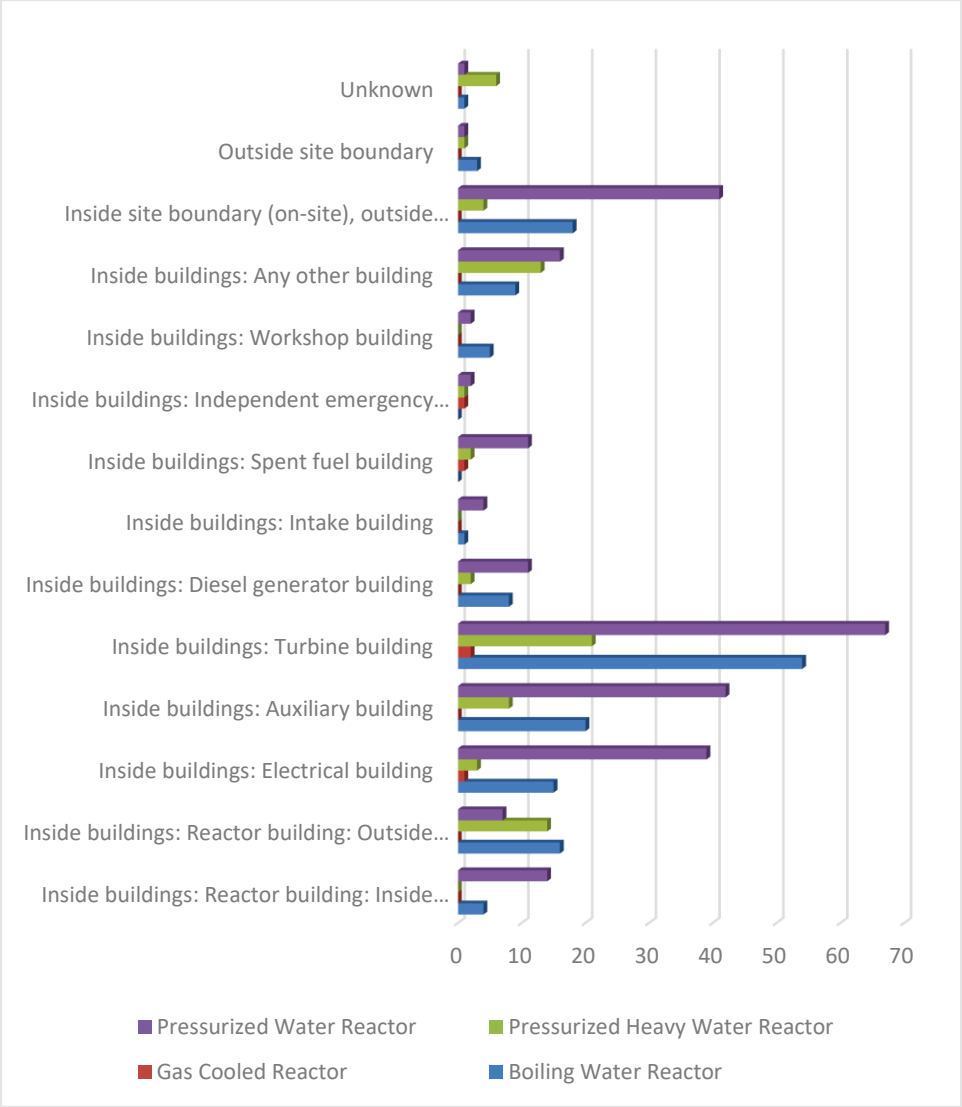


Table 3. Building where fires started [29]

Building Name	BWR	GCR	PHWR	PWR
Reactor building: Inside containment	4	0	0	14
Reactor building: Outside containment	16	0	14	7
Electrical building	15	1	3	39
Auxiliary building	20	0	8	42
Turbine building	54	2	21	67
Diesel generator building	8	0	2	11
Intake building	1	0	0	4
Spent fuel building	0	1	2	11
Independent emergency building	0	1	1	2
Workshop building	5	0	0	2
Any other building	9	0	13	16
Inside site boundary (on-site), outside buildings	18	0	4	41
Outside site boundary	3	0	1	1
Unknown	1	0	6	1

Table 3 and Figure 6 list buildings where fires started for the four types of nuclear reactors. As shown in the table, the “Turbine Building” has the most events for all four types of reactors. The large amount of fires in the turbine buildings result from oil igniting on hot equipment (e.g. bearings, pipes). The turbine buildings are not broken down into smaller areas because there is a large variability among the plants on how these buildings are compartmentalized. Auxiliary buildings on the other hand is the second main building contributor for fires. Many of the fires involved in both the turbine buildings and auxiliary buildings are caused from transformers fires. These fires can be originated at high or medium voltage electrical transformers. The “Spent Fuel Building” and “Independent emergency building” have no fire events for BWR. For PHWR, the “Reactor Building - Inside containment”, “Workshop building” and the “Intake Building” have no fire events as well. However, for the PWR, the building with the least fires starting is the “Outside site boundary” with 1 fire. For GCR, the “Electrical building”, “Spent fuel building” and “Independent emergency building” all have 1 reported fire.

Table 4. Room where fire started [29]

Room	BWR	GCR	PHWR	PWR
Battery room	2	0	1	0
Cable spreading room	0	0	1	1
Diesel generator room	9	0	0	6
Elevator shaft	0	0	0	0
Hydrogen cylinder bunker	0	0	0	0
MCR	1	0	0	3
Not inside NPP building	21	0	5	42
Office	1	0	3	2
Other cable room	1	0	0	4
Other type of room	20	0	9	47
Process room	65	3	19	63
Room for electrical control equipment	12	0	6	23
Room for off-gas equipment	1	0	1	1
Room for ventilation	4	0	4	20
Staircases and/or corridors	1	0	3	3
Storage for combustibles	0	0	0	4
Storage for nuclear waste	1	0	1	1
Storage for other waste	0	0	0	3
Switchgear room	7	2	2	29
Transformer room/bunker	0	0	0	0
Unknown	3	0	16	3
Workshop (controlled area)	5	0	4	3

Table 4 and Figure 7 demonstrate rooms where fires started for all four types of nuclear reactors. The first two locations are the control room and the cable spreading room. These two areas are typically found in every nuclear power plant. Switchgear rooms and rooms for other electrical control equipment providing the highest room specific contributions to fire risk. One of main contributor for fires in process rooms are pumps, where they most likely to reside in such rooms. As illustrated in the table, the "Process Room" has the highest percentage for fires starting for all four types of reactors. While, for BWR, the "Cable spreading room", "Elevator shaft ", "Hydrogen cylinder bunker, "Transformer room/bunker ", "Storage of waste" and "Storage of combustibles" have no fires. For PHWR, which also has no fires in the "Elevator shaft ", "Switchyard", the "Hydrogen cylinder bunker", the "Storage of waste", "MCR", the "Diesel generator room", the "Other cable room", "Storage for other waste", "Transformer room/bunker" and the "Storage of combustibles". On the other hand, PWR has no fires in the following fire zones: "Battery room", "Elevator shaft", "Hydrogen cylinder bunker", "Storage of nuclear waste", "Off Gas room", "Hydrogen cylinder bunker" and "Transformer room/bunker". While for GCR, there were two fires in the "Switchgear room".

Figure 7. Room where fire started [29]

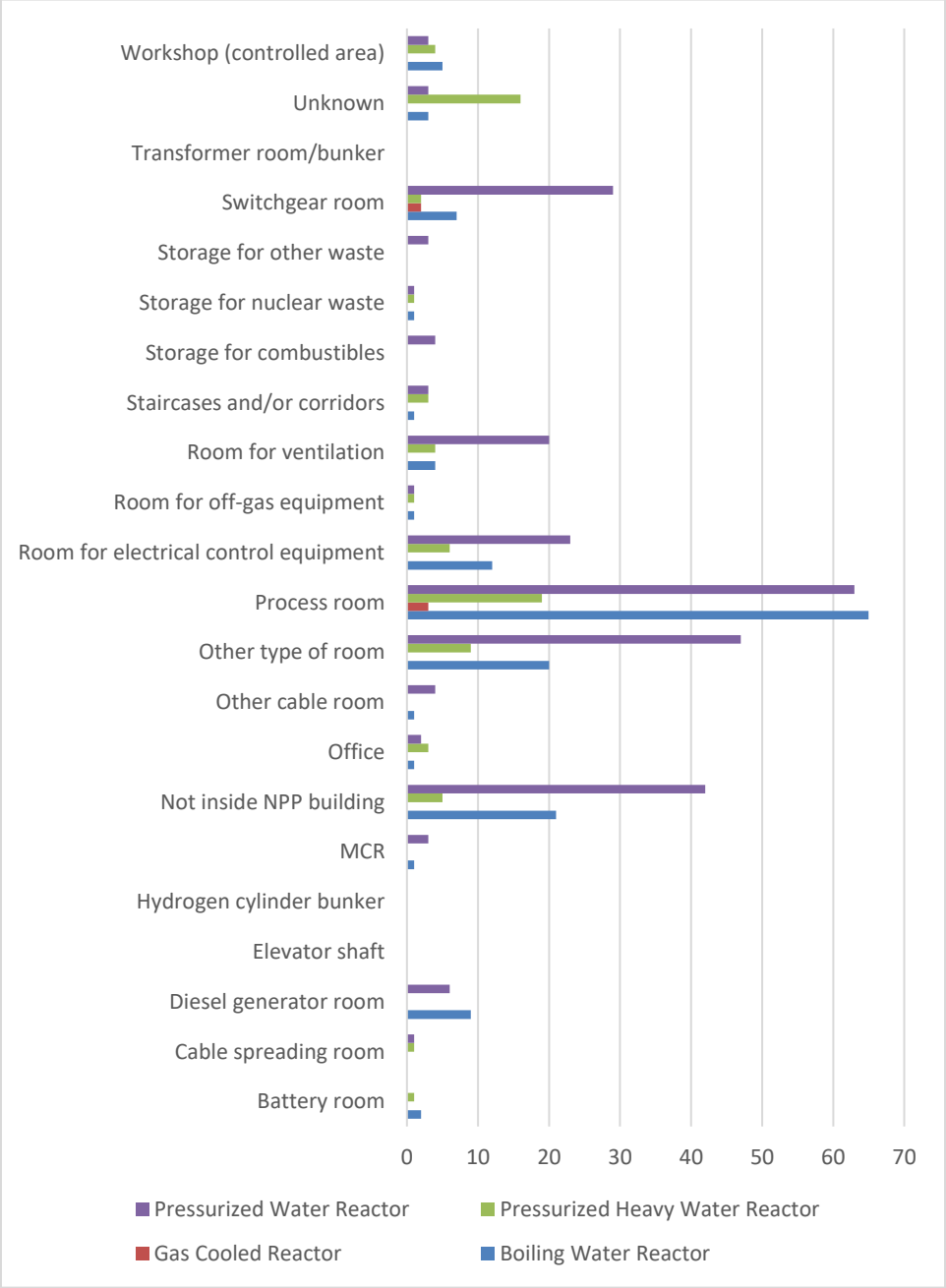


Table 5. Cause of Ignition [29]

Cause of Ignition	BWR	GCR	PHWR	PWR
Electrical	71	1	31	126
Hot component	29	0	15	46
Hot work	20	1	10	35
Mechanical	13	2	9	25
Other	4	1	3	7
Self-ignition	10	0	2	13
Unknown	7	0	5	6

Table 5 and Figure 8 show the causes of ignitions for all three types of nuclear reactors (BWR, PHWR & PWR). Electrical fires are mainly caused by the operation of equipment with the absence of cooling devices, and/or over-operating the equipment than the designed capacity. Mechanical fires are caused mainly by mechanical friction and/or insufficient lubrication that result in overheating. While, hot work fires are caused mainly by sparks that ignite combustibles, transient material and filters. As demonstrated in the table, “Electrical” is the main cause of ignition for all three types of reactors. While, for GCR. “Mechanical” ignition is the main cause of ignition. Whereas, the “Other” cause of ignition has the lowest percentages for BWR, PHWR and PWR. The “Other” cause of ignition simply refers to all other causes other than the ones listed in the table “Hot Component”, “Self-Ignition”, “Electrical”, “Mechanical” and “Hot Work”, such as: “Bursting Pipes”, “Human Accidental Ignition”, etc. For GCR, “Hot component”, “Self-ignition” and “Unknown” has zero fires.

Figure 8. Cause of Ignition [29]

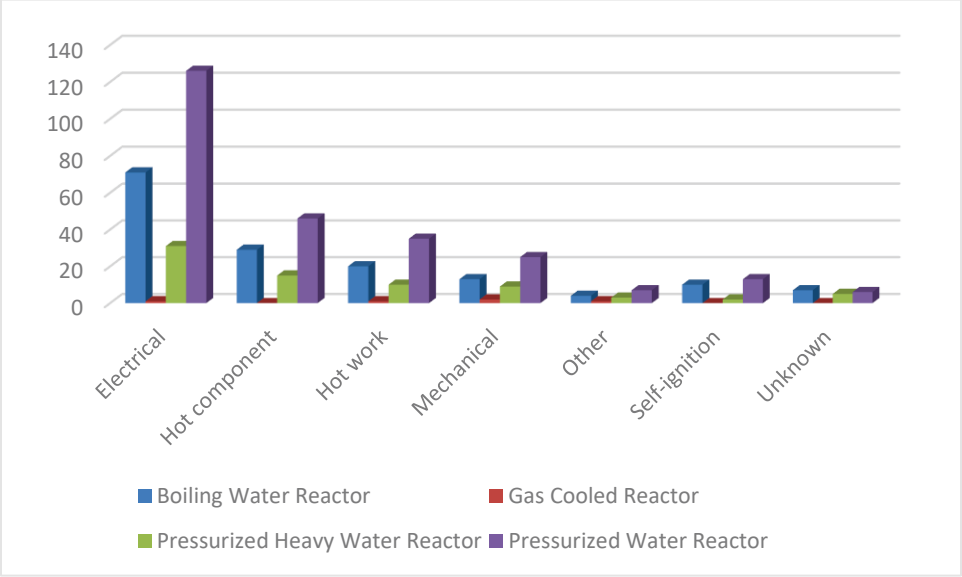


Table 6. Detection system performance [29]

Detection	**BWR**	**GCR**	**PHWR**	**PWR**
Fire detectors were not involved	17	1	22	46
Malfunction	0	0	0	6
No fire detector system in place	18	2	25	52
Normal	114	2	12	141
Unknown	5	0	16	7

Table 6 and Figure 10 show the detection system performance for BWR, GCR, PWHR and PWR of nuclear reactors. Smoke and heat detectors detected most of the fires when detection is available. Plant personnel detection becomes very important in the absence of detection equipment, especially in the events of hot work fires. Plant personnel can quickly extinguish these hot work fires. The highest percentage for the detection performance is “Normal” for both BWR and PWR. While “No fire detector

system in place" is the highest percentage for detection performance for PHWR and GCR. The lowest cause of all four nuclear reactors types is "Malfunction" detection system performance. "Normal" means operated as intended. "Accident detection not involved" means that the criteria for automatic actuation of fire detectors were not met. Detectors were not actuated (correct performance) but would have worked as intended (e.g. in case of low amount of smoke, low temperatures, etc.), fire was detected by other means (e.g. indirect signals, humans being present). "Not involved" refers to fires that were too small to involve the activation of the detection.

Figure 9. Detection system performance [29]

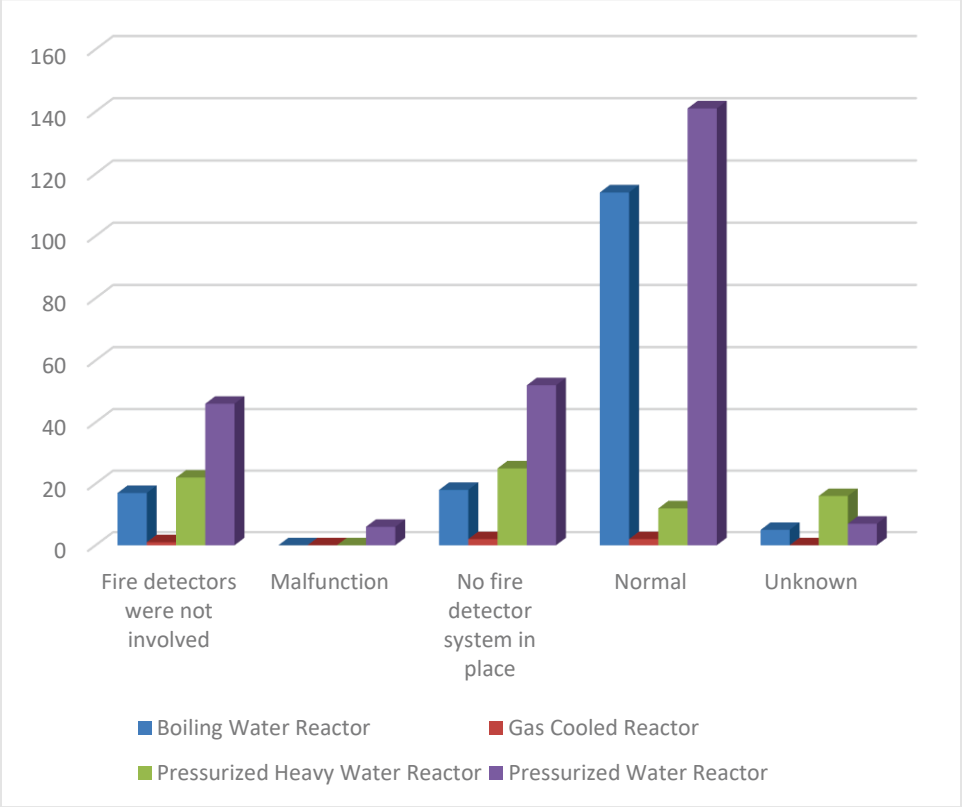


Table 7. Fire extinguishing system performance [29]

Fire extinguishing	**BWR**	**GCR**	**PHWR**	**PWR**
Actuation as intended	19	1	3	39
Malfunction	1	0	0	1
No actuation	70	1	28	108
No fixed extinguishing system present	53	3	36	103
Unknown	11	0	8	7

Table 7 and Figure 10 show the fire extinguishing system performance for all four types of nuclear reactors. In most of the cases, fixed fire extinguishing systems were automatically actuated as designed. The highest percentage for the fire extinguishing performance is “No fixed extinguishing system present” for both PHWR and GCR, while “No actuation” is the highest for BWR and PWR. The lowest cause of all four nuclear reactors is “Malfunction” for the manual fire extinguishing performance. “Actuation is intended” refers to fires who the extinguishing system actuated as intended.

Figure 10. Fire extinguishing system performance [29]

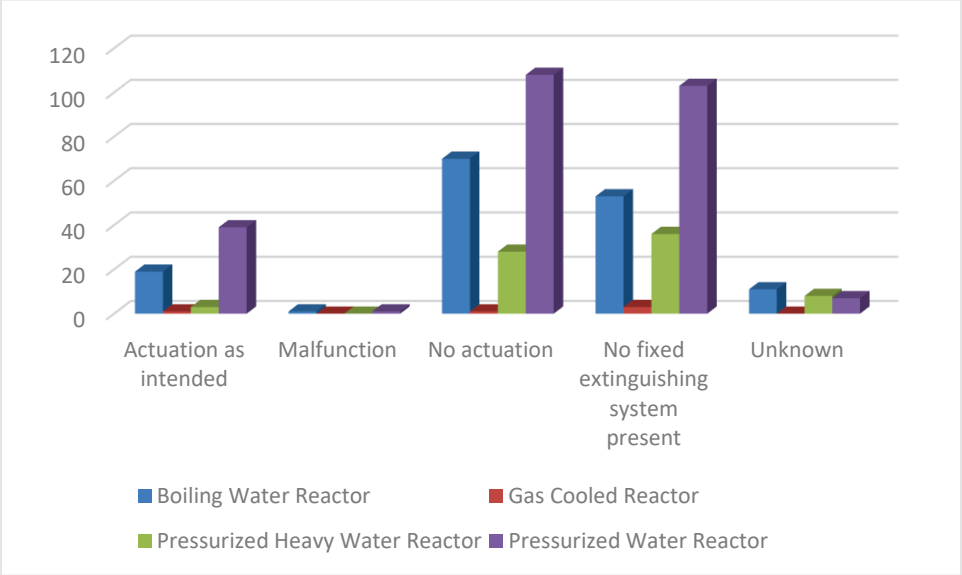


Table 8. Manual firefighting performance [29]

Manual firefighting	**BWR**	**GCR**	**PHWR**	**PWR**
Initial attack successful	90	1	30	118
Not applicable	22	1	30	66
Several attacks needed	31	2	7	68
Unknown	11	1	7	6

Table 8 and Figure 11 show the manual firefighting performance for all four types of nuclear reactors. Many of the fires were extinguished by manual actions. Successful fire extinguishing by manual actions includes but not limited to the use of water hoses, dry chemicals and carbon dioxide. The “Initial attack successful” manual firefighting performance is highest for BWR, PHWR and PWR. While, “Several attacks needed” is the highest for GCR. The lowest manual firefighting performance is “Unknown” for all four types of nuclear reactors.

Figure 11. Manual firefighting performance [29]

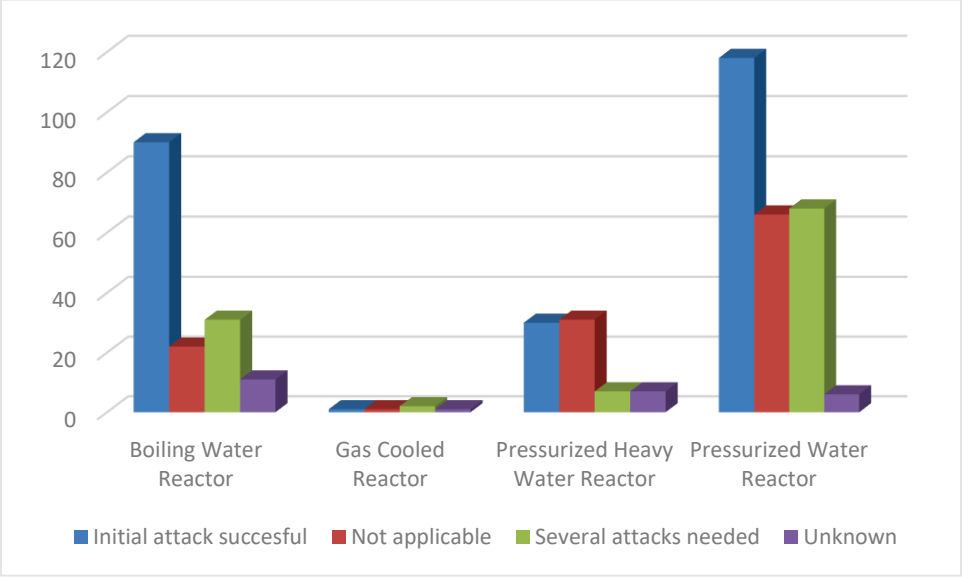


Table 9. Impact to Safety Trains [29]

Safety Trains	**BWR**	**GCR**	**PHWR**	**PWR**
All safety trains affected	0	0	1	5
More than one safety train affected	0	1	0	4
No safety trains affected	131	4	69	210
One safety train affected	20	0	2	37
Unknown	3	0	3	2

Table 9 and Figure 12 show the impact of the fire on safety trains for all four types of nuclear reactors. The “No safety train affected” in the course of fires is the highest number for all four types of reactors during fires. While “More than one safety train affected” due to fire events has the lowest numbers for PHWR. For BWR and GCR, there was no fire that affected all safety trains. For PWR, “Unknown” was the lowest number. Safety trains refers to sets of plant components that perform a safety function.

Figure 12. Impact to Safety Trains [29]

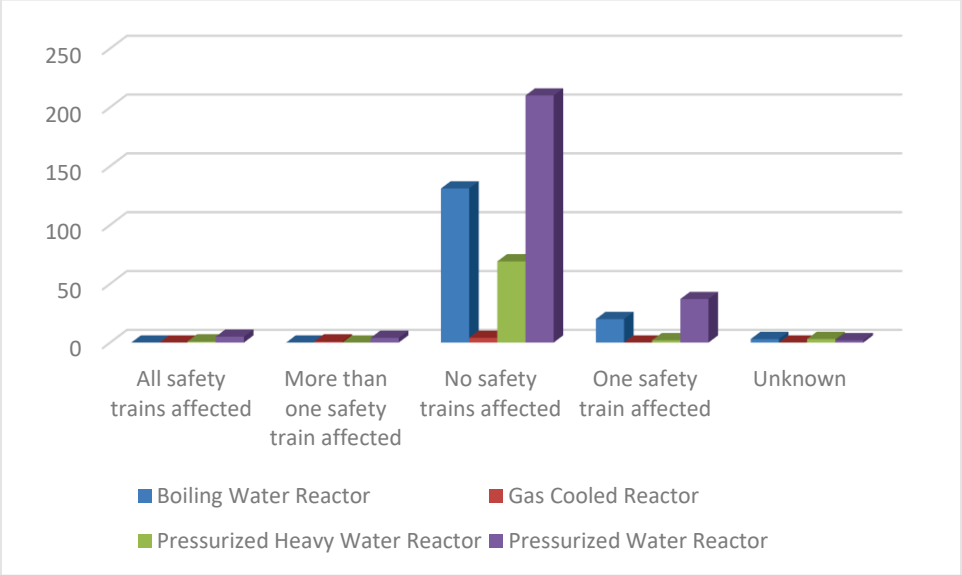


Table 10 shows the number of reported fires in Canada with the corresponding years. Table 10 also illustrates that between 1999 and 2005 had higher reported fire frequencies than the other years. However, it is worth mentioning that there were different fire-reporting criteria prior to 1981. From 1981 to 1996, the fire reporting criteria for each CANDU licensee were listed in their license condition handbooks, and from 1996 to 2003, the fire reporting criteria followed R-99: Reporting Requirements for Operating Nuclear Power Facilities [33]. From 2003 to 2014, the fire reporting criteria followed S-99 [34]: Reporting Requirements for Operating Nuclear Power Plants and from 2014 to today, the fire reporting criteria have followed REGDOC-3.1.1. "Reporting Requirements for Nuclear Power Plants (NPPs)" document [35]. There is a clear relationship between the increased numbers of fires reported since the transition from listing the fire reporting criteria in regulatory documents from the license condition handbook in 1996.

Table 10. Number of fires reported in Canada in CANDU reactor facilities [29]

Year	Canada
1981	0
1982	0
1983	2
1984	0
1985	1
1986	1
1987	0
1988	1
1989	1
1990	0
1991	1
1992	0
1993	1
1994	0
1995	0
1996	0
1997	0
1998	0
1999	1
2000	19
2001	20
2002	8
2003	9
2004	3
2005	5
2006	1
2007	0
2008	0
2009	0
2010	0
2011	0
2012	1
2013	2
2014	0
2015	0
2016	0
2017	0
Total	76

There was an increase in reported fires from 1999 to 2005. The first CSA-N293 standard, Fire Protection for CANDU NPPs, was developed and approved for publication in February 1997 [36]. The CSA-N293 standard was then adopted in the Canadian NPPs' license condition handbooks. CSA-N293 provides the minimum fire protection requirements for the design, construction, commissioning, operation, and decommissioning of CANDU NPPs, including structures, systems, and components (SSCs) that directly support the plant and the protected areas. CSA-N293 also states the fire protection requirements for Canadian NPPs.

Some of the CSA-N293 standard requirements include having fire protection programs and performing fire protection assessments (Code Compliance Review, Fire Hazard Analysis, and FSSA). CSA-N293 also defines the design and installation requirements of fire protection systems. CSA-N293 implementation requires a lot of effort, personnel, and expertise, and requires time to adopt and mature. This could be an explanation for the decrease in the number of fires after 2005, Shalabi, H., and Hadjisophocleous, G., [37, 38].

The two-stage Bayesian model or superpopulation [39] [40] model is to be calculated using the application of Bayes Theorem to plant–specific failure rates population. A modified version [41] was used for calculating fire protection systems and equipment failure rates. The two-stage Bayesian model is summarized below:

Specific emphasis is required to quantify the uncertainties when estimating reliability parameters (such as failure rates). Bayesian approach is used to calculate the quantification uncertainty of failure rate λ. Assume 'k failures observed in cumulated operating time T' the distribution of the failure rate λ. $p(k, T\,|\lambda)$ is the likelihood-function

which in case of failure rate is a Poisson distribution and $p0\ (\lambda)$ is the prior information about the failure rate λ. Assuming a non-informative prior with $p0\ (\lambda) \propto \lambda^{-0.5}$ will result in a Gamma-distribution Γ(α,β) with parameter α = k + 0.5 and β = T [42].

"The expected value of a variable X which is distributed according to $\Gamma\ (\alpha, \beta)$ is given as $E\ (X) = \alpha/\beta$. For that reason the expected value of the failure rate estimation is given as $E\ (\lambda) = (k + 0.5)/T$. This approach yields valid results even in case of zero observed failures in operation time T. The a-posteriori distribution $(\lambda|\ data)$ resulting from the Bayesian approach reflects our state of knowledge of (epistemic) uncertainty according to the estimated failure rate" [42].

"The component operating experience in plant i, is given as (ki, Ti), $i = 1\ldots n$, with ki representing the number of failures observed during operation time Ti in plant i. Assuming the component in each plant $i = 1\ldots n$, has its individual failure rate λi (depending on specific technical characteristics in each plant) which is assumed to have a fixed real value but which is uncertain. The number of failures ki in operating time Ti are realizations of random variates which are Poisson distributed with parameter λi, $I = 1\ldots n$, i.e." [42].

$$p(\mathrm{k}_i, \mathrm{T}_i|\ \lambda_i) = \frac{(\lambda_i \cdot \mathrm{T}_i)^{\mathrm{k}_i}}{\mathrm{k}_{i!}} \cdot \mathrm{e}^{-\lambda_i \cdot \mathrm{T}_i} \quad i = 1, \ldots, n$$ **(Equation 4)** [42]

λi are assumed to be realizations of a random variable Λ which is distributed according to a super population which is assumed to be a Gamma distribution with unknown parameter α and β.

$$\mathrm{g}(\lambda\ |\alpha, \beta) = \frac{\beta^{\alpha}}{\Gamma(\alpha)} \cdot \lambda^{\alpha-1} \cdot \mathrm{e}^{-\beta \cdot \lambda}$$ **(Equation 5)**

The prior of the parameter α and β is derived from Jeffrey's rule [39] and is proportional to $p_0(\alpha,\beta) \propto \alpha^{-0.5}\beta^{-1}$, If $\lambda 1 \ldots \lambda n$ were known, Bayes Theorem would provide:

$$h(\alpha,\beta \,/\, \lambda_1,\ldots,\lambda_n) = \frac{\prod_{i=1}^{n} g(\lambda_i \backslash \alpha,\beta)\cdot p_{0(\alpha,\beta)}}{\int_0^\infty \int_0^\infty \prod_{i=1}^{n} g(\lambda_i \backslash \alpha,\beta)\cdot p_{0(\alpha,\beta)} d\alpha d\beta}$$ **(Equation 6)**

With $p0\ (\lambda i) \propto \lambda i$-0.5 we get

$$\tilde{p}(\lambda_1,\ldots,\lambda_n \mid k_1,T_1;\ \ldots;\ k_n,T_n) = \prod_{i=1}^{n} \frac{p(k_i,T_i \mid \lambda_i)\cdot p_0(\lambda_i)}{\int_0^\infty p(k_i,T_i \mid \lambda_i)\cdot p_0(\lambda_i)\ d\lambda_i} = \prod_{i=1}^{n} g(\lambda_i \mid k_i+0.5,T_i)$$

(Equation 7)

Integrating the product of Equations (5) and (7) over α and β yields

$$\check{p}(\lambda \mid \lambda_1,\ldots,\lambda_n) = \int_0^\infty \int_0^\infty g(\lambda \mid \alpha,\beta)\ \cdot\ h(\alpha,\beta \mid \lambda_1,\ldots,\lambda_n)\ d\beta\ d\alpha$$ **(Equation 8)**

The integration over β can be solved analytically. For integration over α a definite invertible transformation for $\alpha \in (0,\ \infty)$ to $x \in (0,\ 1)$ was used with $\alpha = \phi(x)$. For the transformation the function (x) = - $(x2)$ / $ln\ (x)$ was applied

Using Equations (7) and (8) we derive by applying Bayes Theorem:

$$\hat{p}(\lambda \mid k_1,T_1;\ \ldots;\ k_n,T_n) = \frac{\int_0^\infty \ldots \int_0^\infty \check{p}(\lambda \mid \lambda_1,\ldots,\lambda_n)\cdot \tilde{p}(\lambda_1,\ldots,\lambda_n \mid k_1,T_1;\ \ldots;\ k_n,T_n) d\lambda_1 \ldots d\lambda_n}{\int_0^\infty \int_0^\infty \ldots \int_0^\infty \check{p}(\lambda \mid \lambda 1,\ldots,\lambda n)\ \cdot\ \tilde{p}(\lambda 1,\ldots,\lambda n \mid k1,T1;\ \ldots;\ kn,Tn)\ d\lambda\ d\lambda 1 \ldots d\lambda n}$$

(Equation 9)

The multiple integral in Equation 7, with respect to λii = 1… n is solved by Monte-Carlo simulation. $\hat{p}(\lambda \mid k1, T1;\ \ldots;\ kn, Tn)$ is the generic failure rate distribution calculated by using information of comparable plants. The q% quantiles xq of the generic distribution

are determined numerically by solving the equation $\int_0^{xq} \hat{p}(\lambda \mid k1, T1\ ;\ \ldots\ ;\ kn, Tn\ )\ d\lambda\ =\ q/100$.

The two-stage Bayesian model or the superpopulation model uses Equation 9 as shown in this thesis, which is widely accepted in Canada and internationally for Nuclear PSA calculations [43, 44].

Using Equation 9, Canadian CANDU reactor observations: There are 64 CANDU fires in Canada in 547.90 years. There are 12 non-CANDU fires in Canada in 4.24 years. There are 2 more CANDU fires reported to the FIRE Database outside of Canada in 100.95 years.

Results: Canadian CANDU-specific distribution

- 5%-Quantile: 9.12 E-02/reactor year
- 50%-Quantile: 1.191 E-01/reactor year
- 95%-Quantile: 1.432 E-01/reactor year

The resulting frequency (expected value) for a fire to occur in a CANDU reactor in Canada is 1.19 E-01 per reactor year.

3.1 Conclusion:

This chapter discussed the number of fires per county and per reactor type. The chapter also showed the operation mode prior to fire, the operational mode during fire initiation, the building where fire started, and the room where fire started**.** Additionally, the chapter explored the causes of ignition, detection system performance, fire extinguishing system performance, manual firefighting performance and impact to safety

trains for all types of reactors. Using the two-stage Bayesian model or super population model, the generic fire occurrence frequency of CANDU reactors in Canada is estimated to be 1.19 E-01 per reactor year.

Chapter 4: CANDU Fire PRA Steps

The evaluation of the applicability of NUREG/CR-6850 to CANDU reactors must include an assessment of the adequacy of using the fire safe shutdown analysis list of credited structures, systems, and components (when prepared in accordance with the requirements of CSA N293-12) for use in a Fire PRA (when prepared in accordance with the requirements of NUREG/CR-6850).

4.1 General Approach for a Fire PRA

Fire PRAs can evaluate three levels of risk. The first-level PRA evaluates the frequency of accidents that cause damage to the nuclear reactor core. The Level 2 PRA starts with the Level 1 core damage accidents and assesses the frequency of accidents that release radioactivity from the nuclear power plant, and the Level 3 PRA starts with the Level 2 radioactivity release accidents and evaluates the consequences in terms of injury to the public and damage to the environment.

An inclusive Level 1 Fire PRA should be completed for power operation as well as shutdown plant modes. The Level 1 Fire PRA divides the plant to (n) separate fire zones for the plant operational states and estimates the frequencies of fire-induced damage states per reactor per year (ry). The total frequency of core damage fire-induced (FCDF) is the total of the FCDF for all compartments and plant modes, including full power (FP) as well as low power (LP) and shutdown (SD) states [45].

The German regulatory body has used a threshold value of 1.0 E-07 fires/yr [44] for the Fire PRA for full power modes. First, each fire compartment has been analyzed with respect to fire-specific aspects. If this analysis gives the result that no fires inadmissibly impairing nuclear safety can occur under the boundary conditions of the

plant mode being analyzed, the compartment has been excluded from further analysis for this mode.

4.2 Experience from France in Applying Fire PRAs

The French Nuclear Safety and Radioprotection Institute (IRSN) has developed Level 1 PSAs for NPPs to establish their own opinion on the assumptions and results of the licensee (EDF) PSAs. The IRSN developed a Level 1 Fire PSA for 1,300 MW reactors during power operation. Their objectives included providing their own model to evaluate the results and assumptions of the licensee Fire PSA. The study [48] represents an extension of the IRSN in-house 1,300 MW NPPs Level 1 PSA for internal events. The advancement of a Fire PSA is required because of the significance of fire with respect to the risk of core damage. France has many nuclear power plants (consisting of 58 current plants with 1 under construction). A majority of them were built by the same manufacturer (Framatome) and are operated by the same licensee (EDF).

The IRSN Fire PSA is a limited study, as only the most important fire zones are included. The study includes the failure of components and equipment including cables due to fire, and the impact of fire on operator actions. The IRSN also made some advancements in the R&D area, such as research on damage temperature criteria and on the impact of smoke on component failures. The fire simulations were conducted using the IRSN SYLVIA fire model, which is a two-zone fire model used for determining the development of the temperature in fire compartments. Adjacent fire compartments, in case of fire spreading, are included to assess when the components could fail (failure time). PSAs represent important aspects of the Periodic Safety Review (PSR) for French 1,300 MWe nuclear power plants. It is crucial to stress that at the beginning of the

development of a Fire PSA, conservative hypotheses and parameter values with uncertainties are used for component failures or cable faults. Analysis of the effect of those choices on the PSA results and the identification of possible needs for R&D are crucial [48].

The general approach for conducting the IRSN Fire PSA has been summarized in the following steps. The first step is the selection of the critical fire zones, the second step is the quantification of fire scenario frequencies, where the fire modelling and the quantification of the c fire scenario take place. The last step is to quantify core damage states induced by a fire.

The main lessons learned from The French experience with adopting the Fire PRA methodology are:

- The number of steps in the Fire PRA can be reduced to 3 steps,
- It is important to collect the national operating experience covering all fire events during about 1400 reactor years, and to utilize these data to estimate fire frequencies and to define fire scenarios,
- Fire modeling can be used to simulate fire scenarios and to assess the potential for component failures in the initial and adjacent fire zones, and
- Fire PSA should focus on the most critical safety equipment during fire scenarios in terms of the fire related risks.

4.3 Experience from Germany in Applying Fire PRAs

Nuclear power plants in Germany have been developed and designed in various generations, which is one reason for existing differences in the design of fire protection styles. Due to this, it is crucial to evaluate the current fire safety status of nuclear power

plants and to see if the level of safety remains acceptable. In Germany, NPP safety concepts and licensing regulations by the authorities are mainly based on deterministic safety principles. These principles include prevention and control of abnormal plant operation conditions and incidents by principles such as passive barriers, redundancy and diversity of safety systems in order to ensure safety. Fire PSA is an obligatory part of the PSA in the frame of Periodic Safety Reviews (PSR). They are recognized as an additional tool to the deterministic fire protection assessment in order to support decision-making. Furthermore, validated fire simulation models and codes must be used for deterministic fire hazard analysis (FHA) and probabilistic fire risk assessment. The procedure must be implemented e.g. for modifications related to fire protection. As an example, in the more recent past, a German NPP licensee requested the regulatory body to approve technical plant modifications concerning the spent fuel pool (SFP) cooling. This request made a Fire PRA necessary- Fire Protection remains a topic of importance to safety, even under the given situation that by end-2022, all NPPs will no longer be in commercial operation, due to the high relevance of fires during longer duration post-commercial safe shutdown phase with fuel elements in the SFP and the long decommissioning phase of at least more than ten years with a lot of hot work.

It is necessary to establish Fire PSA event and fault trees related to specific characteristics and boundary conditions for potential fire scenarios. This helps the analysts to calculate the corresponding branch point probabilities and end states. This is done in order to determine fire-induced core damage frequencies (CDFs) or fuel damage frequencies (FDFs). Reliability data with regard to fire protection means are necessary in the context of modeling plant-specific fire event trees. The most recent German PSA

Guide covering this issue was published in 2005 [49]. The respective Supplements on PSA methods and data [50 – 52] provide a lot of technical guidance for performing Fire PRA including recent technical reliability data for different active fire protection features [52].

The German data originated from specific evaluations of operating experiences from various nuclear power plants. The following components and systems were analyzed in six German plants in order to update the reliability data (and in particular, failure rates per hours of plant operation):

- Fire detection systems with the corresponding main fire alarm panels, subsidiary fire alarm boards, detection drawers, detection lines/groups, and automatic and manual fire detectors;
- Fire and smoke extraction dampers in ventilation ducts with different actuation mechanisms (thermally by fusible link or remote controlled typically by electro-mechanical or pneumatic actuation);
- Fire doors between rooms, partly equipped with electrically operated hold-open devices;
- Stationary fire extinguishing systems and equipment, including the corresponding extinguishing supplies, fire pumps, hydrants, etc.

Reliability data were derived from the analysis of files from periodic in-service inspections and additional reports from these findings, and fault trees are provided for more complex systems like fire detection systems. These trees were provided in order to assess the reliability of these systems. The present database was developed to include around 111 plant operational years of six units of various types of nuclear power plants.

In 2009, a state-of-the-art Fire PSA methodology has been developed for German NPP. The Fire PSA methodology is built on multi-step qualitative and quantitative screening approaches, applying a database specifically developed for the application in the frame of Fire PRA [53].

Detailed requirements for fire protection in nuclear power plants in Germany have been updated in 2015, covering basic requirements specified in the safety standard KTA 2101.1 [54]. Requirements for structural elements´ fire protection are provided in the safety standard KTA 2101.2 [55], and requirements for mechanical and electrical components´ fire protection are provided in the safety standard KTA 2101.3 [56]. All three parts of the safety standard KTA 2101 need to be considered in the design and modification of German nuclear powers plants with respect to fire safety. Moreover, these standards need to be considered as design basis requirements when performing Fire PRA.

Some lessons learned from the German experience are:

- It is important to provide accurate data on technical reliability of active fire protection features in various types of NPPs of different plant generations.
- The ventilation conditions to be considered is a requirement in the German fire protection standards KTA 2101, Part 1-3 [54-56].
- The OECD FIRE Database can be used to estimate generic fire frequencies when utilizing the operating experience from fires in nuclear installations

4.4 Finland's Experience in Applying Fire PRAs

Finland's Olkiluoto 1 and 2 NPPs' Fire PRA were updated in 2011 [57] and contain an evaluation of the fire frequency for each component group by applying a Bayesian approach with NUREG/CR-6850 data and using historical fire events occurred the NPPs as evidence. The fire frequency estimations changed for some rooms by orders of magnitude because of the transition to the use of component-based fire frequencies, in particular the fire frequency estimations of rooms with exceptionally high fire frequencies. These rooms are used for keeping components such as water pumps or large amounts of electrical equipment, like relay rooms and rooms with hydrogen systems.

The analysis included mapping safety-related components to their locations, and the cable routing database was used to determine the power and control cables for safety-related components. In this analysis, the locations of cables transmitting data to the reactor protection system were also evaluated. In the old method that is based on NUREG/CR-0654 [58], the fire frequencies of the rooms were normalized so that the estimated total fire frequencies equalled the historical frequencies at Olkiluoto 1 and 2, but in the new method, the evaluations included data from other plants as well. The new method based on NUREG/CR-6850 for estimating fire frequencies increased the estimated total fire frequency of Olkiluoto 1 and 2 by 10 % when compared to the total fire frequency estimate of the unit using the old method. In addition, room-specific fire frequencies changed significantly for some rooms, and the same applies to rooms with the highest fire frequency estimates [57].

The lessons learned from Finland's experience are:

- The need for creating a fire database to estimate fire frequency, and

- Using cable fire loads as fuel in fire scenarios as it dominates the fire risk of many commercial power plants.

4.5 South Korean UCN 5, 6 Fire PSA Model Based on the ANS Fire PRA Standard

The American Nuclear Society (ANS) Fire PRA standard consists of 13 items, as per Table 11 below. The individual items involve both high-level requirements and supporting requirements. The scope of a PRA covered by this standard is limited to analyzing accident sequences initiated by fire that might occur while a nuclear power plant is at nominal full power.

Table 11. Main requirements of 13 elements in the ANS Fire PRA standard [59]

No.	Element	Requirements
1	Plant Partitioning	- Fire area/compartment definitions - Partitioning process - Walk-down and documentation
2	Equipment Selection	- Process of selecting equipments - Location information
3	Fire PRA Cable Selection	- Cable location information - Documentation about cable selection
4	Qualitative Screening	- Qualitative screening criteria - Insignificant area screening
5	Fire Scenario Development and Fire Modeling	- requirements associated with the identification and analysis of these fire scenarios, including the application of the fire modeling
6	Ignition Frequency	- Prohibition on assigning zero ignition frequency, using non-nuclear data - Assuming that a transient combustible fire may occur at any area of the plant
7	Quantitative Screening	- Qualitative screening criteria based on the cumulative risk impact of screened compartments on total LERF and CDF
8	Structures and Fire Modeling	- Requirements for the unscreened area/compartments which has not screened in the quantitative screening process
9	Circuit Failures	- Refinement of the understanding and treatment of fire induced circuit failures on an individual fire scenario basis
10	HRA	- Identification and quantification of events representing human failures
11	Seismic Fire	- Effects of a seismic event on the fire related issues - Four seismic-fire interaction issues presented in Fire Risk Scoping Study
12	Fire Risk Quantification	- Requirements that the quantified risk measures are to include the total CDF and LERF
13	Uncertainty and Sensitivity	- identification and treatment of uncertainties for each portion of the fire PRA

According to the results of the quality evaluation of the Ulchin Nuclear Power Units (UCN 5&6) using the fire PSA model based on the ANS standard, it is impractical to apply the whole NUREG/CR-6850 to the domestic PSA due to the environmental differences between Korea and USA [59].

Lessons learned from the Korean experience:

- The whole NUREG/CR-6850 cannot be applied to the domestic PSA due to the operational and regulatory differences between Korea and USA.
- Performing computational fire modeling and circuit failure analysis as a part of the Fire PRA as it is not used in the Korean's domestic Fire PRA.
- Including the effect of a seismic event on the fire related issues as a part of the Fire PRA as it is not used in the Korean's domestic Fire PRA.
- The Fire PRA should consider human action sin detail.

4.6 Implementation of Internal Fire Probabilistic Risk Assessment (IFPRA) in Japan

Internal Level 1 Fire Probabilistic Risk Assessment (IFPRA) among the PRA for light water reactors at power operation was deliberated by the Fire PRA subcommittee consisting of experts in the related areas under the Risk Technical Committee for the Standards Committee of the Atomic Energy Society of Japan. This standard contributed significantly to PRA engineers for executing the IFPRA with satisfactory quality, recognizing vulnerability regarding internal fires and contributed to further improvement of the NPPs' safety. The IFPRA standard is broken down as follows, Shalabi, H., and Hadjisophocleous, G., [60] see Figures 13, 14, and 15:

（1）Scope and applicability

（2） Cited standards

（3） Glossary

（4） IFPRA procedure and assurance of quality

（5） Collection of plant information

（6） Analysis of plant information

（7） Screening of fire zones

（8） Developing fire scenarios for quantitative screening

（9） Screening of fire scenarios

（10） Developing fire scenarios for detailed analysis

（11） Quantification of accident sequences

（12） Documentation

Figure 13. IFPRA Standard [61]

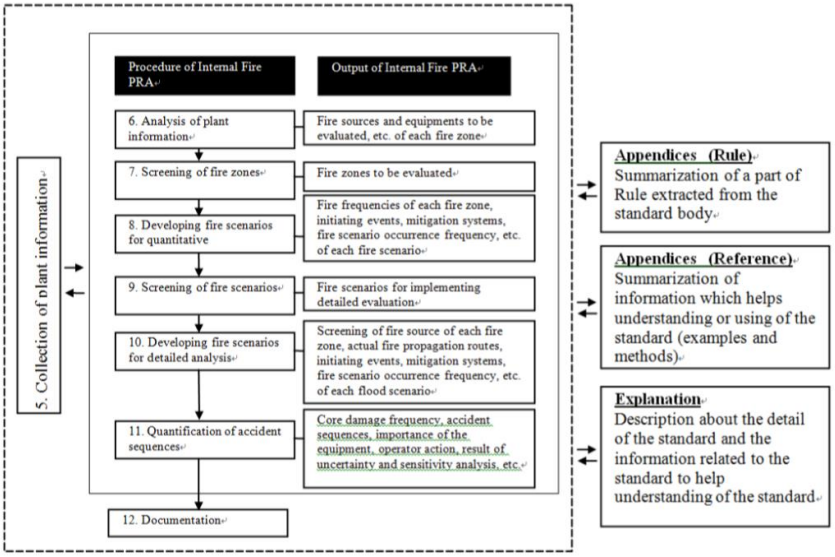


Figure 14. Developing fire scenarios for quantitative screening [62]

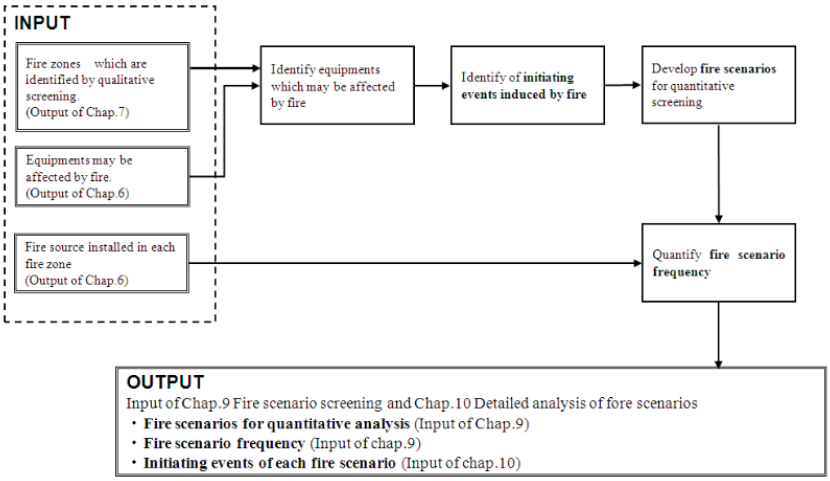
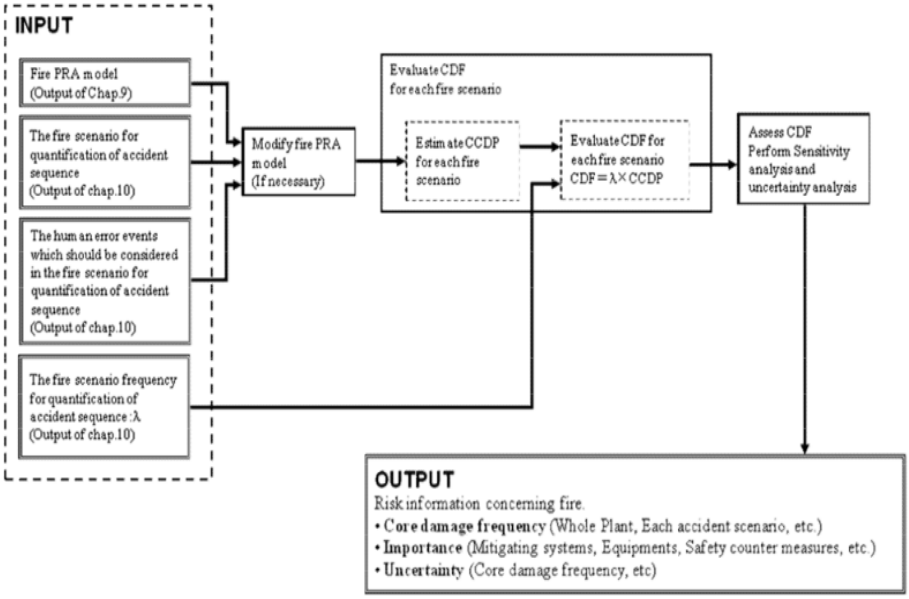


Figure 15. Quantification of accident sequences [62]



The lessons learned from the Japanese experience are:

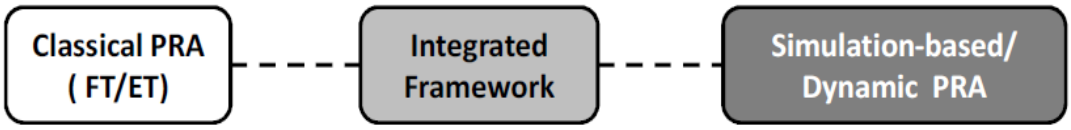
- The number of steps from NUREG/CR-6850 can be reduced.
- The OECD fire Database can be used to estimate the fire frequency to analyze operating experience.

- Fire PSA should focus on the most critical safety equipment during fire scenarios in terms of fire related risks.

4.7 Integrated Fire PRA Framework

For nuclear plants utilizing classical PRAs to shift to fully simulation-based PRAs requires substantial time and resources, and so an integrated PRA modeling framework was proposed in [61] that stands between the classical and simulation-based/dynamic PRAs (see Figure 16). An integrated framework is a "realistic" modeling of fire that could decrease unrequired conservatisms. There are three major features for the Integrated framework: Plant-specific PRA Module composed of ET/FT used in the classical plant-specific PRA, the Fire Simulation Module (FSM) which includes the simulation and uncertainty quantification of realistic phenomenological models for fire initiation, dynamic progression of fire effects, and postfire damage and, the Input which provides the required input for the Fire Simulation Module. A simulation-based / dynamic PRA uses realistic physical models for fire initiation, dynamic progression of fire effects, and post-fire failure by modeling time-dependent fire phenomena and damage.

Figure 16. Modeling the integrated framework with respect to PRA evolution [63]



The objective is not to interpret the fire phenomena into a fault tree (FT) and event tree (ET) context, but to model them in a simulation-based environment to interface with the plant-specific PRA.

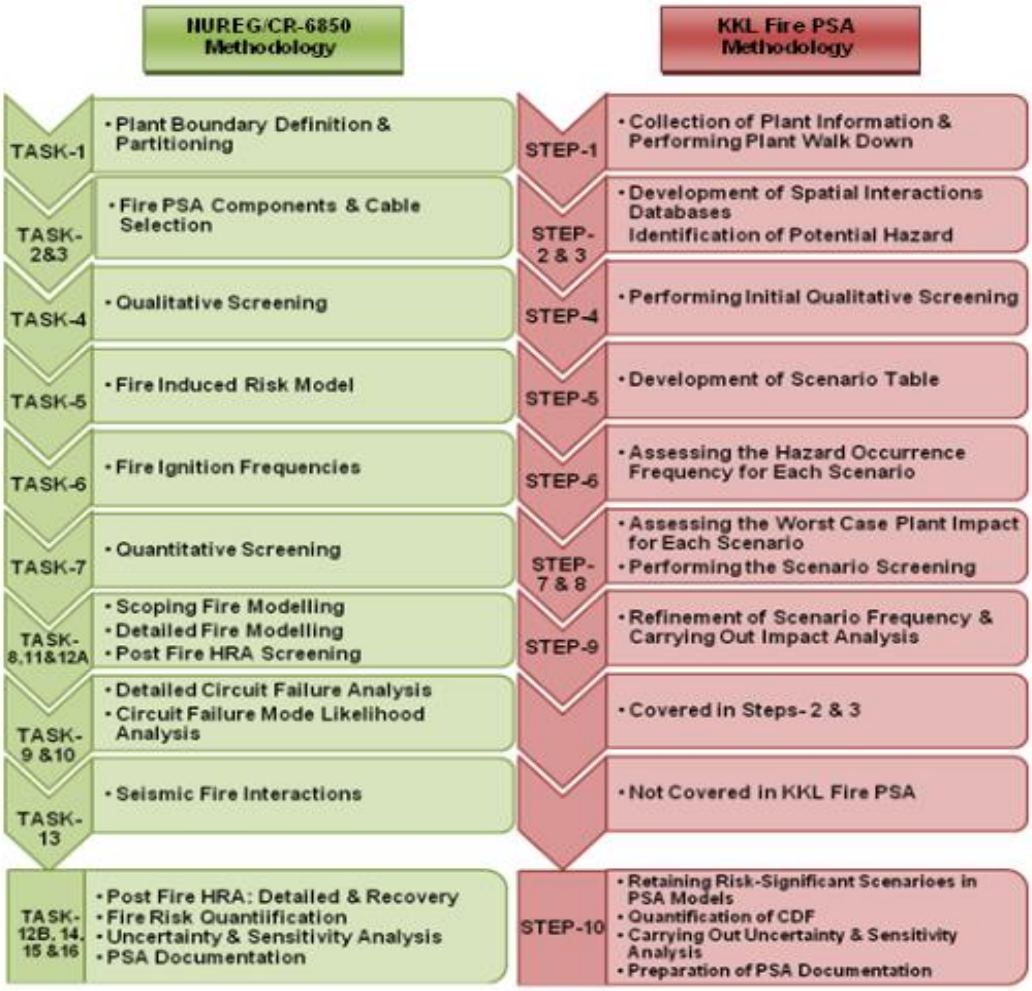
One known disadvantage of the current Fire PRA methodology is the limitation on accounting for dynamic aspects of fire phenomena. As a solution to this limitation, an integrated framework is offered to model the dynamic phenomena to a totally simulation-based or dynamic PRA.

4.8 The Leibstadt Nuclear Power Station Fire PRA Model

Between 2005 and 2009, the Leibstadt Nuclear Power Station (Kernkraftwerk Leibstadt, i.e., KKL) in Switzerland conducted an improvement of its original internal Fire PRA based on the guidance in NUREG/CR-6850, integrating important methodological developments and enlarged modeling features in order to improve the PRA application (see Figure 17).

As noted previously, NUREG/CR-6850 has been criticized with regard to its comprehensiveness and its implementation difficulties in practical applications. Thus, based on the experience of the Leibstadt Fire PRA analysis team, a few changes regarding the NUREG/CR-6850 guidance were made in some areas in order to improve the efficiency of the analysis process. These changes were only made after careful thoughts in order to guarantee that the fundamental technical approach and principal recommendations of NUREG/CR-6850 were not compromised [64]. In Figure 17 the NUREG/CR-6850 tasks are shown in green and the new developed Leibstadt Fire Risk Methodologies (KKL Fire PSA methodology) tasks are shown in red.

Figure 17. Comparison of NUREG/CR-6850 and Leibstadt Fire Risk Methodologies [64]



The Leibstadt internal fire analysis was performed in two phases [64]:

1. Spatial Interactions Analysis, covering: (1) Plant boundary definition and partitioning, collection of plant information, and conducting an extended plant walk-down; (2) development of spatial interactions databases; (3) identification of potential hazard locations and sources; (4) performance of preliminary screening; and (5) development of scenario tables.
2. Detailed Scenario Analysis, covering: (6) Assessment of hazard occurrence frequency for each scenario, partly automated application of the plant database;

(7) assessment of worst-case plant impacts for each scenario; (8) quantitative scenario screening; (9) refinement of scenarios; and (10) retention of risk-significant scenarios in the plant PSA models.

The lessons learned from Leibstadt Nuclear Power Station experience are:

- The number of steps from NUREG/CR-6850 can be reduced from 16 to 10 steps.
- The Fire PRA can be divided into two major parts: Spatial Interactions Analysis and Detailed Scenario Analysis.
- There are two different screening steps, step 4 (performing initial qualitative screening) and steps 7 and 8 (performing the scnerio screening) in theKKL method.
- Task 13 (seismic-fire interactions) in NUREG/CR-6850 is not covered in the KKL method.

4.9 Fire PRA by OECD/NEA/CSNI members update

In the past years, significant discussions on the maturity and realism of fire PSA methods have taken place, with a diversity of views remaining to co-exist on the subject. In 2019 a Topical Opinion Paper (TOP) on Fire PRA by OECD/NEA/CSNI WGRISK members countries has been updated according to the actual state-of-the art [4] clearly stating that Fire PRA is mandatory in several countries as an additional analytical tool and meanwhile represents a mature tool.

4.10 Proposed Model for CANDU Fire PRA, Shalabi, H., and Hadjisophocleous, G., [62]

After reviewing and analyzing the international experience and lessons learned from applying NUREG/CR-6850 in different countries, a CANDU Fire PRA methodology has been developed. The main lessons learned that were taken into consideration while developing the CANDU Fire PRA are minimizing the number of steps from NUREG/CR-6850 since it is impractical to apply the whole NUREG/CR-6850 due to the environmental and plant differences between Canada and the U.S. Creating a qualitative step to reduce the number of fire zones needed to be analyzed in more detail, creating the National operating experience covering all fire events, using fire modeling to simulate fire scenarios and to assess the potential for component failures in the initial fire zone. In addition, using the cable fire loads as fuel in fire scenarios as it dominates the fire risk of many commercial power plants, and including HRA requirements and the effect of combination events on the fire related issues. The main tasks of the CANDU Fire PRA are shown in Table 12.

Table 12. CANDU Fire PRA

Task 1: Plant Boundary Definition and Partitioning
Task 2: Fire PRA Components and Equipment Including Cables
Task 3: Qualitative Analysis
Task 4: Frequency of Fire Starting
Task 5: Circuit Failure Analysis
Task 6: Fire Modeling
Task 7: Post-Fire HRA
Optional Task: Other Combination Hazard(s)-Fire Events
Optional Task: Severe Accident Management
Task 8: Quantification of Fire Risk
Task 9: Uncertainty and Sensitivity Analysis & Documentation

Task 1: Plant Boundary Definition and Partitioning

The purpose of this task is to identify the plant boundaries and partitions in order to divide the plant into fire zones and fire zones. Fire zones are associated with equipment, cable routings, and initiating events, and all boundaries must be clearly documented. The procedures for the first task are summarized as follows:

Step 1: Global plant boundary analysis and partitioning

Step 2: Fire zone information gathering and characterization

Step 3: Documentation

Task 2: Fire PRA Components and Equipment Including Cables

The target components selected will include components that are vulnerable to fires causing thermal effects, smoke, or water spray damage, and unsafe shutdown. The procedures for the second task are summarized as follows:

Step 1: Identify equipment associated with fire-induced initiating events

Step 2: Identify equipment with potential spurious actuations that may challenge safe shutdown capability

Step 3: Identify additional mitigating instrumentation, and diagnostic equipment important for human response

Step 4: Select Fire PRA circuits/cables

Step 5: Develop cable routing and plant locations

Step 6: Fire cable equipment list and target location reports

The Fire PRA Cable List identifies the circuits/cables required to support the appropriate processes of equipment contained in the Fire PRA Equipment List. Important electrical power supplies are also identified during this task. The Fire PRA Cable List is not just a list of cables; it also determines for each cable a link to the related Fire PRA component and to the cable's routing and location. These associations specify the basis for identifying potential equipment failures in fire areas, fire zones, or raceway levels [2].

Task 3: Qualitative Analysis, Shalabi, H., and Hadjisophocleous, G., [65]

This task references the steps that were taken to develop the qualitative step and not introducing fire modelling in an early task. The qualitative screening methodology being introduced here differs from that of NUREG/CR-6850 [2] in Chapter 2 of this thesis, and consists of the following six steps:

- Step 1: Fire Load Survey

  Perform a full fire load survey for each FSSA fire compartment in operating CANDU reactors in Canada.

- Step 2: Area and Height Ranges

  Divide all FSSA fire compartments into categories by area and height. Identify the most common typical areas and heights for all categories.

- Step 3: Combustible Load Ranges

  Divide the combustible loads present in each FSSA fire compartment into categories. Define the most common combustibles present in each category.

- Step 4: Location of FSSA Cables

  Identify the most common distances between combustibles and FSSA cables. Identify the typical location of FSSA cables[4].

- Step 5: Fire Modeling

  Use Computational Fluid Dynamics Models (CFD) to model the worst-case fire scenarios for all defined category combinations (area and height ranges, combustible load ranges, and location of FSSA cables).

- Step 6: Qualitative Screening Decision-Tree Creation

  Based on the FDS results, develop a qualitative decision tree, which can be used to screen in or out fire zones from further quantitative analysis.

Task 4: Frequency of Fires Starting

As shown in Chapter 3 of this thesis, Equation 9 was used to calculate the fire occurrence in CANDU reactor in Canada to be 1.19 E-01 per reactor per year.

[4] This can only be done in CANDU units, since they are so similar, even in cable routing (which is very different from most reactors in the world

Task 5: Circuit Failure Analysis

The purpose of this task is to determine the response of the components in a fire zone so that cables that have no impact on the operation of the component can be filtered out. This task's procedures are summarized in the following steps:

Step 1: Compile and evaluate prerequisite information and data

Step 2: Perform circuit failure mode probability analysis

Step 3: Generate circuit failure mode probability reports

Task 6: Fire Modeling

The Fire PRA approach will use CFD [21] to generate estimates of the target mean time of failure (MTTF) and the target probability of model failure. The process that determines the severity factor (SF) as a function of the source of fire ignition and components and the total probability of failure for a particular component because of the initiators is:

$P_{fail} = P_{SF} * P_{NS}$ (**Equation 10)**

P_{SF} = severity factor

P_{NS} = probability of non-suppression [2]

Task 7: Post-Fire Human Reliability Analysis (HRA)

This task is divided into two parts and includes the quantification of human failure events (HFEs) and the post-fire preparation procedure. The HRA includes:

Step 1: Modifying and adding human failure events to the models

Step 2: Assigning quantitative screening human error probabilities and performing detailed best-estimate analyses of the important HFEs

Step 3: Documenting the (HRA) [2]

Task 8: Quantification of Fire Risk

The quantification of Fire Risk could be done using event trees and/or fault trees illustrating the combinations of internal initiating events. However, for a given initiator fire, many components of the SSD could be the subject of these failures, and the failure of the component will be completely dependent on common causes, e.g., fires caused by an ignition source in a zone fire. Failure due to fire is sometimes referred to as location or environment dependency.

Task 9: Uncertainty and Sensitivity Analysis & Documentation

Calculating the uncertainty of various events used in the model (ignition frequency, severity factors, the probability of non-suppression and human factors) is handled outside the model. The procedures for uncertainty and sensitivity analyses are summarized in the following steps: [2]

Step 1: Identify uncertainties associated with each task

Step 2: Develop strategies for addressing the uncertainties

Step 3: Perform a review of uncertainties and address them [2]

Optional Task: Other Combination Hazard(s)-Fire Events

This task is a qualitative task used to identify potential vulnerabilities or weaknesses and does not require quantitative work. Hazards are defined as events with damage consequences that could involve an entire nuclear facility. Internal hazards are usually initiated inside the nuclear site, while, external hazards are initiated from offsite sources [66]. Hazards can include but are not limited to the following external hazards: seismic, hydrological hazards, biological hazards, and the internal hazards: internal fire, internal flooding, high-energetic component failure (e.g., HEAF), and internal explosion.

Other combination hazard(s)-fire events procedures include the following seven steps, among others:

Step 1: Identify key combination hazard(s)-fire events in fire zones

Step 2: Assess the potential impact of hazard induced fires

Step 3: Assess the c degradation of fire suppression systems and features by the hazard

Step 4: Assess the potential impact of spurious signals of fire detection systems

Step 5: Assess the potential impact of spurious actuation fire suppression systems

Step 6: Assess the potential impact of the hazard event on manual firefighting

Step 7: Complete documentation

Optional Task: Severe Accident Management

In general, severe accident management programmes are designed to:

- Evaluate the capability of existing plants to tolerate and/or mitigate a severe accident;
- Identify events that can lead to severe accidents and formulate preventive and mitigation strategies;
- Identify short- and long-term measures for handling severe accidents [64].

One of the main objectives of this thesis is to develop a methodology for performing Task 3 (Qualitative Analysis). This includes performing a fuel survey and identifying the typical fire zones and simulating these typical fire zones to find out the effect on FSSA cables.

Chapter 5: CANDU Fire Load Densities in Canadian Nuclear Plants, Shalabi, H., and Hadjisophocleous, G., [67]

A fuel survey was carried out in Canadian nuclear power plants for only the fire zones that contain FSSA equipment. CSA N293-12 [27] defines FSSA fire zones are fire compartments that contain FSSA equipment and/or cables where one or more of the FSSA performance goals cannot be met in the case that all of the FSSA equipment and/or cables located in the fire zone are assumed unavailable. FSSA equipment and/or cables are those that are necessary to achieve the following functions:

a) Maintain the integrity of the reactor coolant pressure boundary,
b) Maintain the capability to shut down the reactor and maintain it in a safe condition, and
c) Maintain the capability to prevent or mitigate the consequences of an incident, which could result in potential offsite exposures.

There are 19 operating CANDU reactors in Canada at five sites (Bruce A, Bruce B, Darlington, Pickering, and Point Lepreau). Fire load density surveys were carried out for all FSSA fire zones at all five sites. Data collected from the fire load density surveys included floor areas/ceiling heights, ceiling/wall/floor constructions, available suppression and/or detection systems, accessible fire hoses and/or portable extinguishers, available ventilation and/or penetration, HEAF, all potential ignition sources, and the types and quantities of combustibles.

Switchgears, load centers, and bus bars/ducts can encounter catastrophic failures that manifest as rapid releases of energy in the form of heat and light, and cause severe damage to structures. A failure can lead to the initiation of a fire involving the electrical

device itself or any flammable material that is exposed to it such as cable trays or nearby panels. This type of event is known as HEAF [69]. In this survey, HEAF risk zones are defined as fire compartment that contain specific components such as switchgear, transformers, electric cabinets, cables, connecting boxes, and circuit breakers at voltages larger than 600 Volts.

The following steps were carried out to produce the fuel survey (which were followed at all sites):

1) The FSSA fire zones in all sites were identified and all fire loads in FSSA fire zones were computed and recorded.
2) A general fire zone list for all sites was developed based on the FSSA fire zones' functions.
3) Measuring and determining the mass and/or volume of all types of combustibles as well as their calorific values, the fuel loads were calculated in the fire zones.
4) The maximum fire load and average and maximum fire loads per unit area were determined.
5) The floor areas/ceiling heights, ceiling/wall/floor construction, available suppression and/or detection equipment, accessible fire hoses and/or portable extinguishers, available ventilation and/or penetration, and HEAF were identified.
6) A general list of potential ignition sources and the types of combustibles was generated.

5.1 Results

The fuel survey carried out included 1,230 fire zones. A general fire zone list for all sites was developed to combine fire zones with similar functions, which resulted in 38 general fire zones as shown in Table 13. The fire zones will be grouped up by areas and fire loads and by function as the there are a big range of values between fuel densities. A state-of-the-art fuel inventory was carried out to for all 1,230 fire zones. Grouping fire zones is only a step to figure out medians, mediums for areas and combustible loads to be able to do the less simulation numbers and not meant to represent the FSSA room fuel loads. The following parameters was also analysed in the fuel survey, Shalabi, H., and Hadjisophocleous, G., [67].

- Fire load density: Type and amount of combustibles, compartment size, calorific values, etc.
- Compartment specific fire induced frequency: Compartment specific parameters; e.g. availability of personnel, electrical and mechanical equipment present in the fire zone, etc.
- Probability of fire propagation to an adjacent fire zone: fire detectors and the types of firefighting equipment in the fire zone and adjacent fire zones, shared wall construction type, etc.

Table 13. General fire zones list and functions

	Fire Group Name	**Function**	**Number of Zones**
1	Access Area	Areas providing access to other areas, which contain a variety of equipment from different systems, e.g., a main corridor with MCC panelling contained within the area, access areas to	85

	Fire Group Name	Function	Number of Zones
		different rooms containing gallium stations, valves, etc.	
2	Access Way	A tunnel, lobby, foyer, hallway, stairwell, corridor, elevator, vestibules, roof, etc. These areas do not contain major equipment for plant systems.	153
3	Airlock and Air chamber	Areas which contain airlocks or air chambers	9
4	Annulus Gas	Areas which contain equipment related to the Annulus Gas System	12
5	Battery Room	Areas which contain battery systems	16
6	Boiler Room	Areas containing boiler/boiler-related equipment for multiple systems	32
7	Cable Areas	Areas containing high amounts of cabling (e.g., cable trays running through the entire room), cable spreading areas	37
8	Control Room	Areas which contain main/secondary/safety control rooms	20
9	Decontamination Room	Areas which contain equipment for decontamination of materials; decontamination workshops	10
10	Control Room (ECI)	Areas which contain ECI-related equipment/systems	42
11	Electrical Room	Areas containing a variety of electrical equipment (e.g., switchgear, inverters, MCCs, transformers etc.) for multiple systems	42
12	End Shield Cooling	Areas with equipment for end shield cooling system	4
13	Equipment Area	Areas with a variety of equipment for multiple different systems	87

	Fire Group Name	Function	Number of Zones
14	Emergency Response Team (ERT) Room	Area for ERT equipment, personnel, etc.	5
15	Fuelling Machine	Areas designated for fuelling machines, fuel handling, and FM support systems	24
16	Generator Room	Areas that contain emergency diesel generators, motor generators, etc.	8
17	Heat Transport	Areas primarily containing equipment for the Heat Transport System (e.g., heat exchangers, valves, pumps, boilers, etc.)	64
18	Instrumentation Room	Areas containing instrument equipment	74
19	Moderator Room	Zones associated with moderator systems. This includes heat exchangers, purification systems, D2O vapour recovery systems, D2O circulation pumps, instrumentation, D2O resin systems, D2O collection systems, D2O Tanks, D2O recovery/clean-up areas, etc. Instrumentation Room	81
20	Monitoring Room	Areas which contain equipment for monitoring systems (not included in the electrical room)	23
21	Motor Control	Areas mostly containing MCCs	32
22	Nuclear Storage	Areas with equipment/storage spaces involved with the storage of new or spent fuel (e.g., irradiated fuel bay) and rooms storing equipment to support fuel-storing activities (purification system, liquid handling system, etc.) and any active materials	21
23	Office/ Miscellaneous	Zones that include offices, break rooms, workout rooms, and other general-purpose rooms	24

	Fire Group Name	Function	Number of Zones
24	Pump Room	Areas which contain pumps for miscellaneous systems	4
25	Reactivity Mechanism	Areas with equipment for the reactivity mechanism.	9
26	Reactor Vault	Main reactor vault of unit	13
27	Relay Room	Areas with equipment for relays	4
28	Shutdown	This area includes all zones containing systems/equipment that deal with safety/emergency shutdowns or regular shutdowns (e.g., safety system 1, shutdown cooling, etc.)	28
29	Steam	Areas containing steam-protected rooms	10
30	Storage Area	General storage areas (e.g., tools, office equipment, supplies etc.) and unloading and loading areas	18
31	Switchgear Room	Zones containing switchgear for various systems and equipment (e.g., class III/IV, EPS System, etc.)	29
32	Transformer Room	Areas containing transformers for multiple systems (e.g., pumps, etc.)	17
33	Turbine Room	Turbine halls and turbine auxiliary rooms/bays	76
34	Valve Room	Areas containing valves for multiple systems	10
35	Ventilation and Air Equipment	Areas containing equipment for A/C, exhaust (contaminated/non-contaminated), air systems (e.g., instrument air system)	59
36	Water System	Areas containing equipment for water systems (e.g., ESW, service water, etc.)	35
37	Workshop	General maintenance fabrication work area	9

	Fire Group Name	Function	Number of Zones
38	Zone Control	Areas containing equipment associated with liquid zone control	4

Table 14 illustrates the maximum fuel loads (MJ) and maximum fuel densities (MJ/m^2) for all general fire zones. The maximum fuel load is 2785 GJ, and the maximum fuel load density is 1.3 GJ/m^2. APPENDIX B shows a sample calculation for fuel loads and fuel densities.

Table 14. Maximum and minimum fuel loads and densities

Fire Zone Name	Maximum fuel load (MJ)	Maximum fuel density (MJ/m^2)	Minimum fuel load (MJ)	Minimum fuel density (MJ/m^2)
Access Area	510596	379	2453.5	19.6
Access way	273342	493	74.5	7.2
Airlock & TC	59678	173	552.0	8.7
Annulus Gas	4768	159	862.1	20
Battery Room	49179	1170	5228.2	261.4
Boiler Room	27841	78	200.9	4.3
Cable Areas	783130	770	2660.0	26.4
Control Room	200882	363	4569.0	98.3
Decontamination Room	63229	453	4339.0	96.2
ECI Room	35137	262	319.2	10.6
Electrical Room	55977	215	467.8	23.5
End Shield Cooling	20809	144	1858.0	36.4
Equipment Area	164312	804	1172.5	25.0

Fire Zone Name	Maximum fuel load (MJ)	Maximum fuel density (MJ/m^2)	Minimum fuel load (MJ)	Minimum fuel density (MJ/m^2)
ERT Room	31784	1319	11338.6	200.1
Fuelling Machine	81261	553	1688.6	13.5
Generator Room	209489	946	3609.4	65.6
Heat Transport	215701	793	271.7	2.43
Instrumentation Room	20958	248	74.2	13.5
Moderator Room	136083	946	376.8	20.9
Monitoring Room	16927	172	478.8	19.6
Motor Control	49155	371	1570.0	44.8
Nuclear Storage	100822	214	1184.9	7.1
Office/ Miscellaneous	66260	702	377.1	26.9
Pump Room	54921	215	2567.0	14.1
Reactivity Mechanism	11060	114	2393.9	19.3
Reactor Vault	141866	164	137.4	0.1
Relay Room	6254	58	1986.4	18.6
Shutdown	41845	246	319.2	20.7
Steam	91342	488	2718.0	28.1
Storage Area	494719	524	201.6	28.5
Switchgear Room	559994	860	4468.9	44.0
Transformer Room	494708	697	2427.6	68.9
Turbine Room	2785403	642	715.7	13.7
Valve Room	46063	248	58.8	5.5

Fire Zone Name	Maximum fuel load (MJ)	Maximum fuel density (MJ/m^2)	Minimum fuel load (MJ)	Minimum fuel density (MJ/m^2)
Ventilation & Air Equipment	241168	440	1436.3	14.6
Water System	83641	265	475.5	15.5
Workshop	141380	531	12363.9	44.8
Zone Control	7964	865	2001.8	47.8

Table 15 shows the maximum area, minimum area, average fuel density, and fuel density standard deviation for all 38 general fire zones as per NFPA 557. The average fuel density for all 1,230 fire zones is 170.1 MJ/m^2.

Table 15. Maximum and minimum Areas and average fuel densities

Fire Zone Name	Maximum Area (m^2)	Minimum Area (m^2)	Average fuel density (MJ/m^2)	Fuel Density Standard Deviation
Access Area	1785.0	66.9	120	98
Access Way	1388.0	12.2	130	123
Airlock & TC	344.0	63.3	60	56
Annulus Gas	43.2	16.3	77	47
Battery Room	94.1	18.1	459	296
Boiler Room	447.1	80.0	29	21
Cable Areas	804.6	31.7	249	223
Control Room	724.0	146.5	241	88
Decontamination Room	141.8	27.9	246	126
ECI Room	462.2	11.1	70	52

Fire Zone Name	Maximum Area (m^2)	Minimum Area (m^2)	Average fuel density (MJ/m^2)	Fuel Density Standard Deviation
Electrical Room	392.2	9.5	89	52
End shield cooling	155.2	51.1	91	53
Equipment area	837.2	10.1	205	201
ERT Room	83.6	18.6	637	424
Fuelling machine	502.2	123.0	226	137
Generator Room	196.5	52.8	577	523
Heat Transport	752.5	33.3	387	288
Instrumentation Room	290.7	4.0	57	47
Moderator Room	300.5	28.9	229	207
Monitoring Room	103.1	23.2	59	47
Motor Control	576.6	12.7	149	100
Nuclear Storage	878.9	83.6	78	64
Office/ Miscellaneous	160.4	17.0	223	212
Pump Room	254.9	127.3	109	88
Reactivity Mechanism	84.0	180.5	59	34
Reactor Vault	1106.0	278.0	72	54
Relay Room	107.0	107.0	46	18
Shutdown	232.0	23.0	81	55
Steam	187.0	96.6	171	149
Storage Area	3401.9	9.3	146	139
Switchgear Room	282.4	51.3	219	197

Fire Zone Name	Maximum Area (m^2)	Minimum Area (m^2)	Average fuel density (MJ/m^2)	Fuel Density Standard Deviation
Transformer Room	1320.2	707.9	416	153
Turbine Room	5635.8	687.1	206	205
Valve Room	329.2	6.1	91	84
Ventilation & Air Equipment	3402.1	38.2	129	121
Water System	1423.3	19.2	102	73
Workshop	1041.6	58.1	217	145
Zone Control	41.9	9.2	504	424

Table 16 shows the percentage of fire zones that are equipped with suppression and detection equipment, fire hose inside the fire zone, and portable extinguisher inside the fire zone.

Table 16. Suppression, detection, fire hose and portable extinguisher percentages

	Fire Zone Name	Suppression (%)	Detection (%)	Fire Hose (%)	Portable Extinguisher (%)
1	Access Area	0.0	15.3	69.4	72.9
2	Access Way	3.9	27.5	61.4	74.5
3	Airlock & TC	0.0	11.1	22.2	33.3
4	Annulus Gas	0.0	25	58.3	66.7
5	Battery Room	0.0	31.3	87.5	100.0
6	Boiler Room	0.0	0.0	56.3	56.3
7	Cable Areas	5.4	75.7	48.6	37.8

	Fire Zone Name	Suppression (%)	Detection (%)	Fire Hose (%)	Portable Extinguisher (%)
8	Control Room	10.0	95.0	35.0	90.0
9	Decontamination Room	0.0	0.0	50.0	60.0
10	ECI Room	2.4	11.9	54.8	61.9
11	Electrical Room	0.0	95.2	59.5	64.3
12	End Shield Cooling	25.0	100.0	75.0	75.0
13	Equipment Area	4.6	51.7	54.0	70.1
14	ERT Room	0.0	80.0	40.0	60.0
15	Fueling Machine	0.00	50.0	45.8	75.0
16	Generator Room	0.0	100.0	50.0	100.0
17	Heat Transport	0.0	26.6	34.4	51.6
18	Instrumentation Room	0.0	77.0	40.5	44.6
19	Moderator Room	2.5	21.0	58.0	64.2
20	Monitoring Room	0.0	21.7	43.5	47.8
21	Motor Control	9.4	50.0	62.5	78.1
22	Nuclear Storage	42.9	47.6	57.1	66.7
23	Office/ Miscellaneous	16.7	62.5	33.3	58.3
24	Pump Room	0.0	50.0	50.0	100.0
25	Reactivity Mechanism	0.0	0.0	44.4	66.7
26	Reactor Vault	0.0	0.0	30.8	46.2
27	Relay Room	0.0	100.0	75.0	100.0
28	Shutdown	0.0	39.3	53.6	57.1
29	Steam	0.0	50.0	90.0	100.0
30	Storage Area	5.6	16.7	55.6	66.7
31	Switchgear Room	3.4	89.7	79.3	100.0

	Fire Zone Name	Suppression (%)	Detection (%)	Fire Hose (%)	Portable Extinguisher (%)
32	Transformer Room	23.5	0.0	100.0	100.0
33	Turbine Room	40.8	31.6	98.7	100.0
34	Valve Room	0.0	0.0	0.0	70.0
35	Ventilation & Air Equipment	3.4	35.6	35.6	78.0
36	Water System	11.4	37.1	68.6	74.3
37	Workshop	22.2	55.6	66.7	100.0
38	Zone Control	0.0	0.0	0.0	25.0

Table 17 shows the percentage of the fire zones that have mechanical ventilation availability and HEAF risk. The HEAF risk was found in 254 out of the 1,230 general fire zones.

Table 17. Mechanical ventilation and HEAF percentages

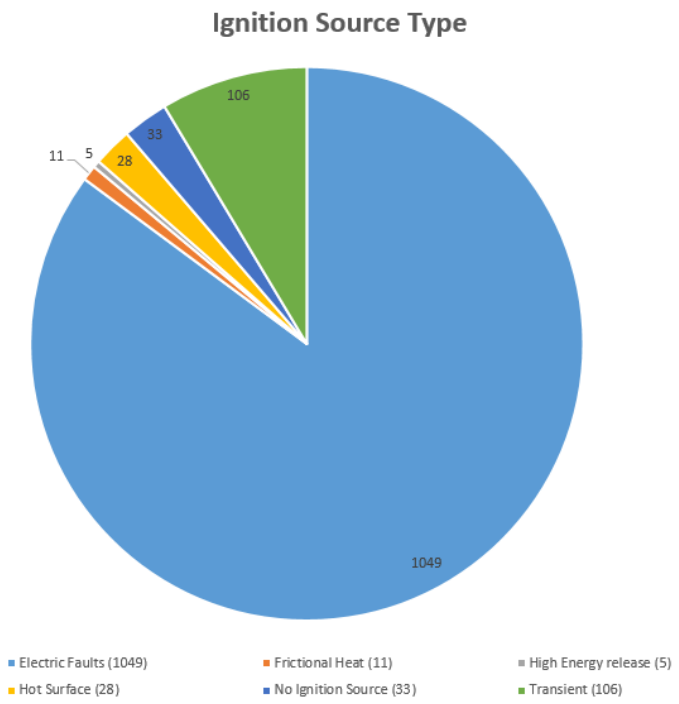
Fire Zone Name	Mechanical Ventilation (%)	HEAF (%)	Ceiling / Walls / Floor
Access Area	23.5	32.9	Concrete
Access way	28.8	11.1	Concrete
Airlock & TC	22.2	0.00	Concrete
Annulus Gas	41.7	8.3	Concrete
Battery Room	0.0	0.0	Concrete
Boiler Room	34.4	0.0	Concrete
Cable Areas	8.1	2.7	Concrete
Control Room	45.0	20.0	Concrete
Decontamination Room	100.0	0.0	Concrete
ECI Room	38.1	7.1	Concrete

Fire Zone Name	Mechanical Ventilation (%)	HEAF (%)	Ceiling / Walls / Floor
Electrical Room	37.3	47.6	Concrete
End Shield Cooling	50.0	0.0	Concrete
Equipment Area	64.4	12.6	Concrete
ERT Room	60.0	0.0	Concrete
Fuelling Machine	66.7	8.3	Concrete
Generator Room	25.0	25.0	Concrete
Heat Transport	60.9	15.6	Concrete
Instrumentation Room	90.5	5.4	Concrete
Moderator Room	40.7	1.2	Concrete
Monitoring Room	52.2	0.0	Concrete
Motor Control	50.0	87.5	Concrete
Nuclear Storage	71.4	4.8	Concrete
Office/ Miscellaneous	58.3	0.0	Concrete
Pump Room	75.0	0.0	Concrete
Reactivity Mechanism	88.9	0.0	Concrete
Reactor Vault	76.9	0.0	Concrete
Relay Room	0.0	100.0	Concrete
Shutdown	46.4	0.0	Concrete
Steam	0.0	60.0	Concrete
Storage Area	55.6	5.6	Concrete
Switchgear Room	3.4	79.3	Concrete
Transformer Room	70.6	88.2	Concrete
Turbine Room	55.3	77.6	Concrete
Valve Room	10.0	0.0	Concrete

Fire Zone Name	Mechanical Ventilation (%)	HEAF (%)	Ceiling / Walls / Floor
Ventilation & Air Equipment	37.3	18.6	Concrete
Water System	25.7	2.9	Concrete
Workshop	88.9	11.1	Concrete
Zone Control	100.0	0.0	Concrete

Figure 18 shows that electric faults were the greatest percentage of ignition sources with 1,049 in 1,230 fire zones, at roughly 85 % of all ignition sources. The electric faults includes but not limited to 600 V transformers, electrical cabinet, electrical panel, electrical switch, fans, junction box, motorized damper, Motor control center (MCC), monitor, printer/fax machines, switches, motors, pump motors, AC units, compressors, steam generators, electric vehicles, motorized valves, relay cabinets, breakers / BUS cabinets, ventilation units, electrical panels, batteries, power supply cabinets, rectifiers and inverters. Figure 18 is based on the ignition sources inventory for all 1,230 fire zones.

Figure 18. Ignition sources chart



Locations of FSSA Cables

The fuel load survey identified that the typical distances from FSSA cables to the nearest combustibles were 2 m, 5 m, or 7 m. In fire zones with an area less than 50 m^2, the maximum distance between FSSA cables and the combustibles was found to be 5 meters. FSSA cable distances to the nearest combustibles are based on the fuel locations during the fuel load survey for the 1,230 fire zones.

5.2 Discussion

The fuel load survey was carried out using the NFPA 557 combination method. The fuel survey was carried out at all operating CANDU reactors in Canada at five sites (Bruce A, Bruce B, Darlington, Pickering, and Point Lepreau) only in fire zones that

contained Fire Safe Shutdown Analysis (FSSA) equipment. The typical distance between combustibles and FSSA cables was found to be 2 m, 5 m or 7 m.

From the fuel survey, it was found that all ceiling, wall, and floor construction in the 1,230 fire zones was made of concrete. FSSA fire zones have different fuel loads. The survey results showed that typical combustibles in these FSSA fire zones include the following: miscellaneous power cables; plastics; cable insulation; lube oil; 2", 3", 4", or 6" diameter electrical cables; compressed gas cylinders; furniture (e.g., desks, chairs, etc.); ½", 1", 2", 2 ½", or 3" diameter rubber hose; plastic tubes; oily waste station; rubber; cardboard; wood; cloth; and trash cans.

A general fire zone list for all sites was developed to combine fire zones with similar functions, and hence 38 general fire zones were identified. All 1,230 fire zones were grouped based on their functionality. Fuel densities and fuel loads were calculated, maximum and minimum areas were identified for all 38 general fire zones. The average fuel density for the 1,230 general fire zones was 170.1 MJ/m^2, and the average fuel load was 79,183 MJ. The maximum fuel load was 2,785,404 MJ, and the maximum fuel load density was 1,319 MJ/m^2.

The potential for electrical faults was found in about 85% of the fire zones. Electrical faults can result from electrical transformers, electrical cabinets and panels, switches, fans and ventilation units, junction boxes, motorized dampers, MCCs, monitors, printers/fax machines, motors, AC units, compressors, steam generators, heat transport pumps, valves, relay cabinets, breakers, batteries, rectifiers, and inverters.

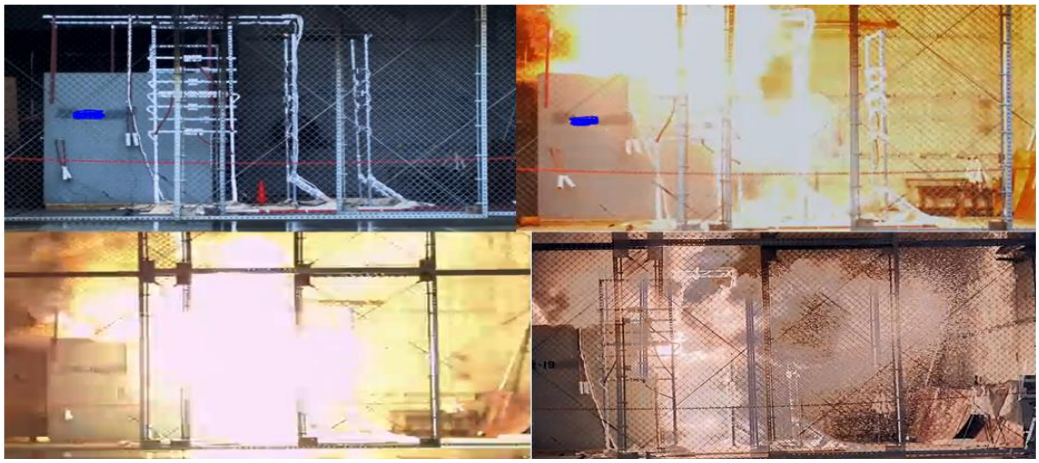
Chapter 6: CANDU High Energy Arcing Faults (HEAFs) in Canadian Nuclear Plants, Shalabi, H., and Hadjisophocleous, G., [68]

In nuclear power plants, switchgear provides a means to isolate and de-energize specific electrical components and buses in order to clear downstream faults, perform routine maintenance, and replace necessary electrical equipment. These protective devices may be categorized by their insulating media, such as air or oil, and are typically specified by voltage classes, i.e., low, medium, and high voltage [69, 70]. Switchgear, load centers, and bus bars/ducts can encounter catastrophic failures that manifest as rapid releases of energy in the form of heat and light and cause severe damage to metal. This type of failure may eventually lead to the initiation of a fire involving the electrical device itself or any flammable material that is exposed to it, such as cable trays or nearby panels. This type of event is known as a High Energy Arcing Fault (HEAF). The event is also referred to as high energy, energetic or explosive electrical arc [69-72].

These high-energetic events most commonly take place in two phases, with the first phase being mainly distinguished by a rapid release of electrical energy. Consequently, a catastrophic failure occurs in electrical enclosures along with the ejection of hot projectiles from damaged electrical components. The first phase of a HEAF may also result in a fire involving electrical devices; while the second phase is characterized by a consequential (ensuing) fire initiated in phase one, which normally results in severe damage of combustible material in the HEAF zone of influence (ZOI). A HEAF in electrical equipment is initiated in one of three ways: a poor physical connection between e.g. the switchgear and the holding rack, environmental conditions, or the introduction of a conductive foreign object.

Figure 19 shows how a HEAF progressed in only a few seconds. The degree of damage includes pressure rise effects (e.g., severe equipment deformation, thrown doors, degraded fire barriers) that can potentially affect equipment in other fire zones. Pressure effects are mainly dependent on room configurations and the electrical characteristics of the event.

Figure 19. HEAF progression from the experiment [73]



Operating experiences from nuclear power plants (NPPs) and other nuclear installations worldwide have showed a non-negligible number of HEAF fire events and resulting fires, Shalabi, H., and Hadjisophocleous, G., [68]. These incidents characteristically occur within high voltage components such as switchgear and circuit breakers or involving high voltage cables. In addition, the numbers of HEAF incidents appear to be increasing. The “Fire Events” report [71] demonstrates that HEAF fire events have the potential to be major risk contributors with substantial safety consequences and considerable economic loss.

In 2009, the OECD Nuclear Energy Agency (NEA) initiated an international investigation of HEAF fire events in NPPs in order to better understand the fire risks at

such plants. The OECD/NEA task explored the effects of HEAF fire events by investigating various phenomena. These phenomena included the effects of enclosure pressures and temperatures, heat release rate, and heat flux on target equipment. Other electrical parameters such as arc voltage, current, and duration were also measured during the experiments. Moreover, the OECD FIRE Database Project indicated that 48 of the total 392 fire events were HEAF-induced fire events [32].

6.1 Discussion: NUREG/CR 6850 Fire PRA Methodology for Nuclear Power Facilities

NUREG/CR 6850 currently provides guidance on how to assess the fire risk of HEAF fire events through probabilistic risk assessment by using both statistical and fire modeling. These methods define the zones of influence and materials ignited by HEAF fire events initiated in switchgear equipment and bus ducts. The main standards used for calculation purposes are NFPA 70E [74] and IEEE 1584 [75]. Computer models are available, most of which are based on the calculations provided by IEEE 1584 or in Appendix D of NFPA 70E.

The NUREG/CR 6850 [2] assessment methodology is based on defining the potential zones of influence for HEAF fire events. A potential zone of influence is defined according to the electrical features of its components. A zone of influence can be investigated to estimate the potential damage to safety-related structures, systems, and components (SSCs). According to NUREG/CR 6850, HEAF fire events are assumed to occur in electrical equipment that operates at 440 V or more. Motor Control Centers (MCCs) with switchgear, which is used to operate equipment, should be evaluated as potential HEAF sources. According to NUREG/CR 6850, switchgear that operates at

greater than 4.16 kV is likely to experience HEAF fire events that will cause impacts outside the cabinet of origin. However, for panels with lower voltage, the effects are estimated to remain within the panels [76].

Vulnerable components or movable/operable structural elements located within 0.9 m horizontally of either the front or rear panels/doors and at or below the top of the faulting cabinet section will suffer physical damage and functional failure [2]. Exposed cables or other exposed flammable or combustible materials or transient fuel materials located within this same region (0.9 m horizontally) will be ignited [2]. In HEAF fire events within the zone of influence of electrical cabinets and switchgear, the initial arcing causes destruction of the faulting device, and adjacent switchgear within the cabinet trips open and the cabinet door blows open due to the metal plasma and mechanical shock. The cabinet fire will burn with high intensity and severity as documented in experiments in NUREG/CR 6850. Unprotected cables will be damaged and drop into the panel in an open-air configuration, and any unprotected cable in the first overhead cable tray, if it is located within a 1.5 m vertical distance from the top of the cabinet will be severely damaged. In addition, any unprotected cable or flammable material located within 0.9 m of the front door of the cabinet will be ignited [75].

There are two methods used for evaluating the effect HEAF. The first is described in NUREG/CR-6850, "Detailed Methodology, Appendix M for Chapter 11, High Energy Arcing Faults" [2], which specifies the following effects:

- The first unprotected cable tray within 1.5 m of the top of the cabinet will ignite.
- Vulnerable equipment within 0.9 m horizontally of the front or rear doors will be destroyed.

- The faulting device will be destroyed.
- Adjacent cubicles in the same cabinet bank will trip open.
- Unprotected cables that enter the panel in an airdrop configuration will be destroyed.

This method has an unidentified degree of uncertainty because it resulted from a single well-documented HEAF event that occurred at the San Onofre Nuclear Generating Station in the U.S. [77].

The second method is in Supplement 1 of NUREG/CR-6850 and EPRI 1011989 Section 7, bus duct (counting) guidance for high-energy arcing faults (FAQ 07-0035) [78]. This method specifies the following effects for HEAF fire events in bus ducts:

- Exposed combustible material within these zones will ignite.
- An ejection of molten material from the bottom of the bus duct and downward to the right in a circular cone at an angle of 15 degrees from the vertical axis. The ejection of molten material will be outwards from the fault point in the shape of a sphere with a 1.5 feet radius.
- A diametrical expansion of the cone will fall straight down in a roughly 20 foot diameter cylindrical shape.

The Core Damage Frequency (CDF) associated with a HEAF scenario involving damage only to the first target is calculated as:

$CDF_i = \lambda_g \cdot W_L \cdot W_{is} \cdot P_{ns} \cdot ccdp_i$ **(Equation 11)**

Where λ_g = the generic frequency for HEAFs,

W_L = the location weighting factor,

W_{is} = the ignition source weighting factor,

$ccdp_i$ = conditional core damage probability (CCDP.

P_{ns} = the probability of non-suppression, if all targets are inside the ZOI, a value of 1.0 should be assumed. If there are postulated targets outside the ZOI, the probability of no suppression can be calculated following the approach provided in Appendix P of the manual suppression curve for HEAFs [78].

6.2 HEAF fire events in Canada

Details on the conclusion of the Canadian's HEAF fire events identified in the OECD FIRE database are as follows:

The first HEAF fire event in Canada occurred in April 2005. An electrical fault and fire on the Main Output Transformer (MOT) caused a turbine to trip. A severe failure occurred on the blue phase winding of the MOT at the same time with a failure of the phase bus potential transformer (surge arrester) cubicle (explosion but no fire). The fire was extinguished by the transformer deluge system. Subsequent to notification of the event, the reactor was manually shut down. The transformer with a capacity to hold 68,000 liters casing was ruptured, and as a result, thousands of liters of transformer insulating oil were spilled over a period of 1-2 hours. Semi-annual oil analysis indicated normal results, and no significant gas increase was noticed until the failure occurred. However, analysis of the root cause indicated that according to the evidence at hand, the incident was caused by a material failure in the MOT. In order to determine the failure mode, forensic disassembly was arranged. As the transformer was unable to dissipate the generated heat, a failure occurred, and the pressure rise caused by the undissipated heat removed the doors from the transformer cubicle, deformed the cubicle, and opened fused links.

The second HEAF event reported in Canada took place in October 2005. A 600 V AC 1600 A Class 4 breaker in a cubical malfunctioned and this led to an electrical arc/fireball. As the breaker was moved to the connect position, an electrical fault occurred, and consequently, a catastrophic failure occurred. The failure led to an electrical arc and fireball that erupted from the switchgear causing burns to an employee. The emergency response team instantly terminated the fire. The employee was moved and diagnosed with first and second degree burns on the thighs. After the fire was terminated, the breaker was repaired; it was tested three times in the test position before receiving permission to be connected again. At the time of the breaker failure, the door of the breaker fire zone was in place and the screws were engaged.

In order to prevent similar failures in the future, corrective actions were taken, and significant improvements were made to protocol and equipment designs. Moreover, a verification process was implemented. Since shutters prevent inadvertent contact between components and buses during maintenance, they play a significant role in the process. Therefore, manufacturers investigated shutter design improvement opportunities. Broken shutters were removed from the breaker cabinet, and staff received applicable training. A team of electrical experts was chosen to address the shortfalls of protective clothes used by the personnel reporting to such events. Lastly, the emergency communication protocol was improved.

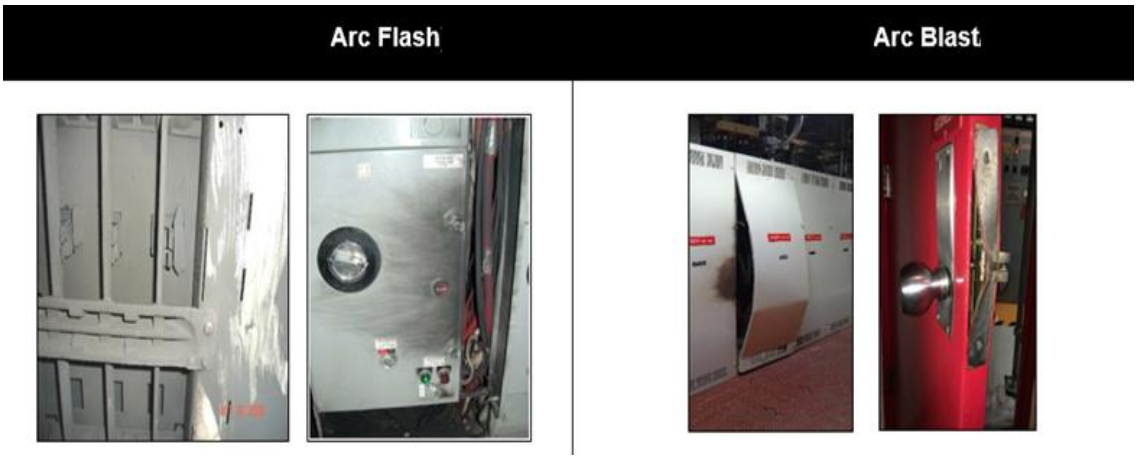
6.3 Arc Flash versus Arc Blast

Arc flash and arc blast phenomena differ vastly from one another. Arc flash is the extremely high-temperature discharge produced by an electrical fault in air. In an arc flash, the damage is contained within the general confines of the component of origin.

These events are associated with minor damage and minimal bus bar degradation from melting/vaporization [79]. The arc blast wave is initially faster than the sound wave. Arc blast is a high-pressure sound wave caused by a sudden arc fault. In an arc blast, damage is contained within the general confines of the component of origin; however, arc blast effects have the potential to damage surrounding equipment through pressure-rise effects (e.g., severe equipment deformation, thrown doors, degraded fire barriers).

During an arc flash, the voltage is likely to drop to approximately 10 % of the original voltage, and the current remains unaffected. In an arc blast, however, both the current and voltage remain unchanged for approximately 1 to 2 seconds, and consequently, the power load of an arc blast is much larger. Therefore, arc blast information is very difficult to obtain in a testing environment without causing power problems. The intensity of an arc flash is thus less than that of an arc blast. Figure 20 shows the differences in damage caused by arc flash and arc blast.

Figure 20. Arc Flash versus Arc Blast



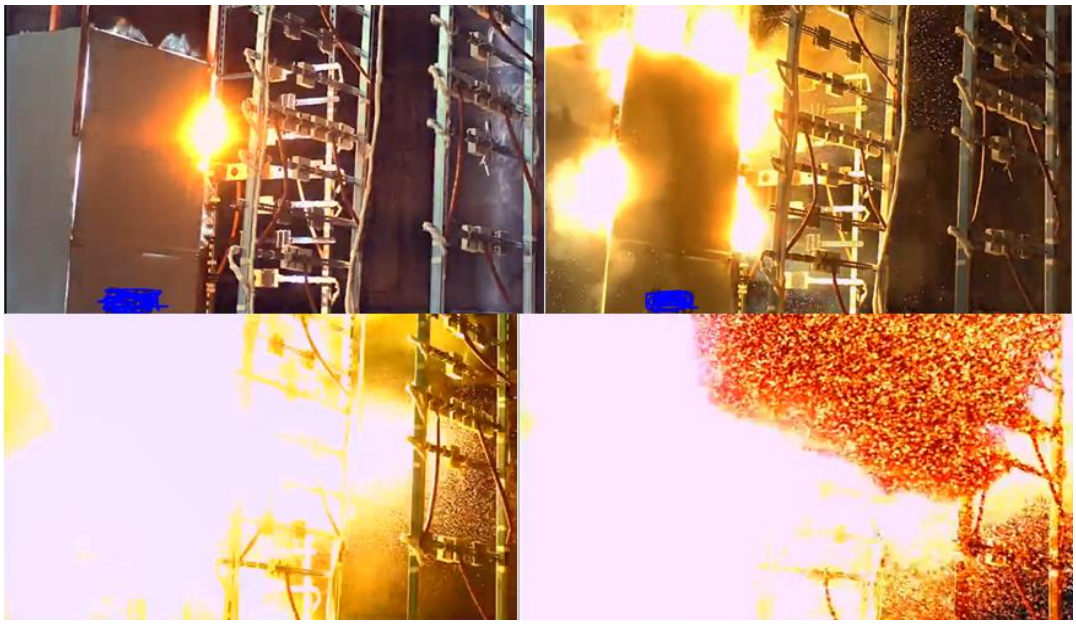
6.4 Copper versus Aluminum

It has been shown that HEAF tests involving aluminum lead to significantly larger energy releases than HEAF tests involving copper. The aluminum in components, subcomponents, or parts of the normal current-carrying pathway could become involved in the fault current pathway because of a ground fault. Aside from the larger release of energy during HEAF fire events when aluminum is involved, dispersal of electrically conductive aluminum by-products throughout the area was observed during real scale HEAF tests carried out within a specific international test program launched by the OECD/NEA called "OECD HEAF" [78]. These by-products were not only conductive but also caused short-circuiting and grounding of equipment in the area. Through the OECD/NEA testing program, when aluminum was involved, the HEAF fire events caused extensive damage to measurement and recording equipment and electrical supplies. According to FAQ Number 07-0035 [80], the damage done by HEAF with aluminum involvement was extensively larger than the damage limit estimated by NUREG/CR 6850. More tests showed a behavior different from NUREG/CR 6850 (even with copper).

A detailed description of the HEAF experimental test series is as follows (from [73]. The tests involved a 4.16 V bus duct section. The bus duct section was removed from a decommissioned U.S. nuclear plant and consisted of non-segregated copper bus bars, which were enclosed in an aluminum duct. The bus duct section was secured to the floor of the enclosure using wooden structural members, and the open ends were secured with electrically insulating fibre. The fibre was utilized to create a pressure boundary inside the bus duct so that it would limit the free energy release once the fault was

initiated. Directly after the HEAF event was initiated, the hot gas mixture pushed the fibre away, which allowed the arc and associated hot gas and plasma mixture to jet out of the bus duct. Consequently, hot gas and molten metal blew from the bus duct opening, as shown in Figure 21. The hot gas and plasma extended nearly 9 m in horizontal direction, causing damage to components in that range; any instrument located in that zone was affected by the high temperatures and heat fluxes.

Figure 21. Aluminum HEAF test [73]



6.5 HEAF Prediction Models

In order to predict the damage to SSCs due to HEAF fire events, simple prediction models have been used based on incident energy models such as the heat release model [2] [76] [78]. The models however are not very accurate, because HEAF events have a very short duration and cause very rapid increases in temperature and pressure due to the release of rapidly expanding superheated vapor, and intense radiation.

Development of an empirical correlation for incident energy and SSC damage is a challenge.

Due to the complications introduced from an energetic electric arc, traditional plume fire models cannot be applied directly to the HEAF energetic phase to measure the heat release rate. A HEAF event might result in a fully developed ensuing fire involving multiple components depending on the exposed materials and equipment involved, while fire plume models can only be applied when the fire is fully established. Based on arc flash calculations, assuming that the incident energy or the energy release rate can be applied using the IEEE 1584 and NFPA 70E calculation methods, two approaches have been proposed.

The first approach assesses [2] the potential for ignition of nearby SSC or equipment by using the point source radiation model. This approach tends to measure the potential for ignition of nearby SSCs by using the Critical Heat Flux (CHF) and then using Thermal Response Parameters (TRP) in order to calculate the estimated time to ignition by using the incident heat flux. In order to determine whether ignition is likely to happen, the estimated time is compared to the arc fault duration time. The applicability of this approach is, however, limited only to situations with arc faults of small durations. Sufficient data was collected and a range of expected arc heat flux intensities generated by the steady current that was established. The incident heat flux is then calculated at various distances by using the same point source radiation model as detailed in NUREG-1805 [81].

The second approach [2] suggests that heat generated in closed switchgear cabinets and MCC panels will develop into fully developed fires either when the energy

dissipated by the fault reaches 10 kW/m^2 or when the energy dissipated by the fault current is in excess of 20 kW/m^2. When any of these mentioned conditions occurs, transformers and bus ducts can develop fires. In these cases, the subsequent fire will affect flammable materials, and a typical plume fire model and point-source ignition models are then employed to measure the impact on nearby objects.

In an effort to understand the characterization of HEAF events, a series of experiments with different arc durations and materials used in the tested components (Al and Cu) was recommended by the OECD/NEA task group on HEAF and further confirmed to be continued after the first HEAF experimental project under the auspices of OECD/NEA had been finished in 2017 Project experiments. These experiments will obtain comprehensive scientific fire data on the HEAF phenomena with potential ensuing fires that are known to occur in nuclear power plants. These experiments will help in developing a more realistic model to account for failure modes and the consequences of HEAF induced fire events in addition to the development of correlations based on ignition times by using variations of incident heat fluxes. These experiments can also be applied to validate current models used to measure SSC damage potential, and moreover, will advance the knowledge and provide better characterizations of HEAFs in the Fire Probabilistic Risk Assessment (Fire PRA) in addition to helping in the prediction of potential damage from HEAF events. In addition, some key questions can be answered by the recommended experiments within the OECD/NEA HEAF 2 Project experiments. Such questions include: how HEAF events can be minimized or prevented, and how signs of HEAF events can be detected [76].

Recommendations for Canadian Nuclear Power Plants

Based on the OECD HEAF Project experiment and testing results, the following high level recommendation list was developed. Specific recommendations should be developed for every Canadian Nuclear plant.

- Review and update maintenance practices designed to eliminate the underlying causes of faults (loose connections, degraded insulation, foreign materials, etc.);
- Limit the magnitude of HEAF fire events so that little damage can occur to nearby equipment and that they do not endanger the functionality of redundant equipment through the design and location of plant SSCs. Identifying the equipment that has potential HEAF fire events and ensure that any redundant equipment is located at a distance larger than 5 m can do this;
- Identify potential weaknesses in the overall plant design to evaluate the benefits of plant modifications for improving safety;
- Conduct an industry survey on the extent of aluminum use in all nuclear stations;
- Revise CSA N293 "Fire Protection for Nuclear Power Plants" so that contents are modified in a risk-informed manner to provide a level of safety commensurate with the risks and probabilities;
- Make use of research results for HEAF and avoid using the guidance provided in NUREG/CR 6850 for ZOIs.

Recommendations for International Nuclear Industry

- Use the experimental results of the OECD/NEA Projects HEAF [70] and the ongoing work on HEAF 2 project [82] results to quantify the ZOIs for HEAF fire events that involve aluminum.

- Revise the current HEAF guidance in NUREG/CR-6850 also recommended by the participants of the HEAF Project.
- Support Fire PSAs by expanding on the "rule of thumb" currently provided in NUREG/CR-6850 by assessing the adequacy of existing HEAF ZOIs in NUREG/CR-6850 for electrical cabinets with aluminum bus bars and for bus ducts containing aluminum.
- Assess the design, effectiveness, and robustness of suppression systems for HEAF induced fires. Additional research is required on the characterization of the enduring fire aspects of such event

Chapter 7: Qualitative Analysis in CANDU Fire PRA, Shalabi, H., and Hadjisophocleous, G., [83, 84]

The following is a detailed description of the work done for the qualitative screening step of the CANDU Fire PRA. The purpose of the qualitative screening step is to "screen in" FSSA fire zones for further quantitative analysis or "screen out" FSSA fire zones from further analysis, as the screened-out fire zones do not represent any damage risk to the FSSA cables. This chapter discuss the approach to perform step number 3 qualitative analysis for CANDU fire analysis. For this approach, the results of the fire load survey are used in order to group the fire zones into other fire zones that have similar areas and fuel loads. Following this grouping, FDS will be used to simulate different fire scenarios. The following will provide detailed description.

As discussed in Chapter 5, a fire load survey was carried out for all operating CANDU reactors in Canada (Bruce A, Bruce B, Darlington, Pickering, and Point Lepreau). The survey resulted in a fire zone group list for all operating CANDU reactor sites in Canada and grouped fire zones by similar functions. 38 fire zone groups resulted from this survey. The results of the survey showed that the average fuel load density for all 1,230 fire zones was 170.1 MJ/m^2 and the average fuel load was 79,183 MJ. High Energy Arcing Faults (HEAF) risk was found in 254 fire zones, and electrical faults were identified as the highest ignition source risk in the 1,230 fire zones, Shalabi, H., and Hadjisophocleous, G., [65].

7.1 Canadian FSSA rooms analysis

To develop a list of FDS simulations, the FSSA fire zones were divided into five groups based on their areas and heights in having the same combustible load, same distance between fire and FSSA cables. Five area groups were selected as shown Table 18: 26 - 50 m^2; 50 – 100 m^2; 100 – 400 m^2, 400 - 1,200 m^2 and over 1,200 m^2. Any selected areas will produce the same linear relationship for temperature vs. area graphs. The area ranges were chosen so that each group will have approximately the same number of fire zones. The number of fire zones in each group, the average and median height and area are shown in Table 18. All fire zones with areas over 1,200 m^2 are screened in for further quantification analysis because the combustible loads are significant and because these fire zones did not follow any regular pattern. In Chapter 5, Table 15 showed the maximum and minimum areas for all 38 fire zones. From Table 15, it can be also seen that there was a wide range between the minimum and maximum areas for all 38 fire zones and therefore, the five area groups in Table 18 can be used to assess different areas for all 38 fire zones. Table 18 shows the averages and the medians for the five area groups. The median values will be used in the simulations as they represent the maximum number of real fire zones.

Table 18. CANDU's FSSA Fire Zone Areas and Heights

Range	Number of zones in range	Average Area (m^2)	Median Area (m^2)	Average Height (m)	Median Height (m)
26 - 50 m^2	335	38.6	38.0	5.4	4.2
50 - 100 m^2	286	74.2	74.0	8.9	4.4
100-400 m^2	370	271.3	187.7	7.0	6.0

400 - 1,200 m^2	155	676.9	571.0	7.2	7.0
> 1,200 m^2	84	2,610.7	1,674.0	9.2	7.2

The purpose of this step is to group the FSSA fire zones into three groups based on their combustible loads. Different simulations were carried out to match the combustible load, area of fire zone and distance between the fire and the FSSA cable to the threshold of 205°C for unqualified cables. It was found that the simulation that had 500 MJ, fire zone area of 26 m^2 and 2 m distance between the FSSA cable and the fire had an output of 183.7°C. In addition, the simulation that had 5,000 MJ, a fire zone area of 400 m^2 and 7 m distance between the FSSA cable and the fire had an output of 181.2°C. Both output temperatures (183.7°C and 181.2°C) are very close the defined threshold of 205°C. Therefore, it was decided to divide the combustible loads into three groups: 0 – 500 MJ; 500 - 5,000 MJ and > 5,000 MJ. The group > 5,000 MJ was screened in for further quantification analysis as it has very significant combustible loads and the consequences will definitely damage the FSSA cables, especially if the entire combustible load is involved in the fire scenario. These three combustible load groups are shown in Table 19 together with the associated number of fire zones, and the typical combustibles.

Table 19. Typical Combustibles in Fire Zones

Combustible load	Number of FSSA Zones	Typical Combustibles (% Combustibles)
0 – 500 MJ	47	Power cables (cable trays) ~ 7.6 m
500 - 5,000 MJ	332	Power cable s(cable trays) ~ 80 m
> 5,000 MJ	853	Power cables (cable trays) ~ 76.2 m (10 %) and plastics ~ 45.5 kg (90 %)

The fuel load survey identified that the typical distances from FSSA cables to the nearest combustibles were 2 m, 5 m, and 7 m as shown in Chapter 5. In fire zones with an area less than 50 m^2, the maximum distance between FSSA cables and the combustibles was found to be 5 m, Shalabi, H., and Hadjisophocleous, G., [65].

In the event of fire, the larger and higher the fire zone the lower the temperature and heat flux because the hot gases mix with more air. The median and average values for the areas and heights are shown in Table 18. In all cases, it was found that the median values for the areas and heights were less than the average values. Therefore, the median values selected and used in the simulations represented more severe scenarios than the average areas. Best-case fire scenarios are the scenarios with maximum areas and heights, while worst-case fire scenarios are the scenarios with minimum areas and heights. Table 20 lists 66 fire scenarios based on the combinations of "area and height", "fire load", and "distance between fire and FSSA cables", Shalabi, H., and Hadjisophocleous, G., [65].

For each fire scenario, the median, the lowest and the highest area for each group was considered for FDS simulation in order to obtain a curve of temperature versus area, which will assist in determining the area that will result in the cable threshold temperature.

Table 20. Full list of FDS simulations

	Fire Scenario number	**Combustible Load (MJ)**	**Area (m^2)**	**Height (m)**	**FSSA cable distance from fire (m)**
1	**1**	~ 500	38.0	4.2	2
2	**1.1**	~ 500	26.0	4.2	2
3	**1.2**	~ 500	50.0	4.2	2
4	**2**	~ 500	38.0	4.2	5
5	**2.1**	~ 500	26.0	4.2	5
6	**2.2**	~ 500	50.0	4.2	5
7	**3**	~ 500	74.0	5.3	2
8	**3.1**	~ 500	50.0	5.3	2

	Fire Scenario number	Combustible Load (MJ)	Area (m^2)	Height (m)	FSSA cable distance from fire (m)
9	**3.2**	~ 500	100.0	5.3	2
10	**4**	~ 500	74.0	5.3	5
11	**4.1**	~ 500	50.0	5.3	5
12	**4.2**	~ 500	100.0	5.3	5
13	**5**	~ 500	74.0	5.3	7
14	**5.1**	~ 500	50.0	5.3	7
15	**5.2**	~ 500	100.0	5.3	7
16	**6**	~ 500	187.7	6.1	2
17	**6.1**	~ 500	100.7	6.1	2
18	**6.2**	~ 500	400.0	6.1	2
19	**7**	~ 500	187.7	6.1	5
20	**7.1**	~ 500	100.0	6.1	5
21	**7.2**	~ 500	400.0	6.1	5
22	**8**	~ 500	187.7	6.1	7
23	**8.1**	~ 500	100.0	6.1	7
24	**8.2**	~ 500	400.0	6.1	7
25	**9**	~ 500	571.0	7.0	2
26	**9.1**	~ 500	400.0	7.0	2
27	**9.2**	~ 500	1200.0	7.0	2
28	**10**	~ 500	571.0	7.0	5
29	**10.1**	~ 500	400.0	7.0	5
30	**10.2**	~ 500	1200.0	7.0	5
31	**11**	~ 500	571.0	7.0	7
32	**11.1**	~ 500	400.0	7.0	7
33	**11.2**	~ 500	1200.0	7.0	7
34	**12**	~ 5,000	38.0	4.2	2
35	**12.1**	~ 5,000	26.0	4.2	2
36	**12.2**	~ 5,000	50.0	4.2	2
37	**13**	~ 5,000	38.0	4.2	5
38	**13.1**	~ 5,000	26.0	4.2	5
39	**13.2**	~ 5,000	50.0	4.2	5
40	**14**	~ 5,000	74.0	5.3	2
41	**14.1**	~ 5,000	50.0	5.3	2
42	**14.2**	~ 5,000	100.0	5.3	2
43	**15**	~ 5,000	74.0	5.3	5
44	**15.1**	~ 5,000	50.0	5.3	5
45	**15.2**	~ 5,000	100.0	5.3	5
46	**16**	~ 5,000	74.0	5.3	7
47	**16.1**	~ 5,000	50.0	5.3	7
48	**16.2**	~ 5,000	100.0	5.3	7
49	**17**	~ 5,000	187.7	6.1	2
50	**17.1**	~ 5,000	100.0	6.1	2
51	**17.2**	~ 5,000	400.0	6.1	2

	Fire Scenario number	Combustible Load (MJ)	Area (m^2)	Height (m)	FSSA cable distance from fire (m)
52	**18**	~ 5,000	187.7	6.1	5
53	**18.1**	~ 5,000	100.0	6.1	5
54	**18.2**	~ 5,000	400.0	6.1	5
55	**19**	~ 5,000	187.7	6.1	7
56	**19.1**	~ 5,000	100.0	6.1	7
57	**19.2**	~ 5,000	400.0	6.1	7
58	**20**	~ 5,000	571.0	7.0	2
59	**20.1**	~ 5,000	400.0	7.0	2
60	**20.2**	~ 5,000	1200.0	7.0	2
61	**21**	~ 5,000	571.0	7.0	5
62	**21.1**	~ 5,000	400.0	7.0	5
63	**21.2**	~ 5,000	1200.0	7.0	5
64	**22**	~ 5,000	571.0	7.0	7
65	**22.1**	~ 5,000	400.0	7.0	7
66	**22.2**	~ 5,000	1200.0	7.0	7

As discussed earlier, simulations will be done for fuel loads of 500 MJ and 5,000 MJ, and using the corresponding areas, heights, and distances between fires and FSSA cables. As the conditions in the fire zones in terms of temperature and heat flux depend on the areas of fire zones, simulations were done using three areas to investigate its impact on the temperature and heat flux to the FSSA cable. Sixty-six fire scenarios have been identified as shown in Table 20. The list of fire scenarios was reduced to twenty-three FDS simulations in Table 21 using the following arguments:

Simulations 1-11 use a combustible load of 500 MJ. From these scenarios, scenarios 1.1, 2.1, 3.1, 6.1 and 9.1 represent the worst-case fire scenarios at a 500 MJ load, for the four different areas and heights at a distance of 2 m between the fire and the FSSA cables. Therefore, if these scenarios' temperature and heat flux output do not exceed the threshold of unqualified and/or qualified cables, then the rest of the 11 simulations would not exceed these thresholds. Scenarios 12-22 use a combustible load of 5,000 MJ. From these scenarios, scenarios 13.2, 16.2, 19.2 and 22.2 have been selected

for simulation because they represent the best-case scenarios at 5,000 MJ. Therefore, if these scenarios' temperature and heat flux output exceed the threshold of unqualified and/or qualified cables, the rest of the 11 simulations will exceed these thresholds as well. On the other hand, if the temperature and heat flux output do not exceed the threshold of unqualified and/or qualified cables, additional scenarios with distances less than 7 meters between the fire and the FSSA cables should be examined.

Table 21. The twenty-three FDS Fire Simulations

Simulation Number	Fire Scenario	Combustible Load (MJ)	Area (m2)	Height (m)	FSSA cable distance from target (m)	Best-case / Worst-case / Median / Area range average
1	**1**	~ 500	38.0	4.2	2	Median
2	**1.1**	~ 500	26.0	4.2	2	Worst-case
3	**1.2**	~ 500	50.0	4.2	2	Best-case
4	**2**	~ 500	38.0	4.2	5	Median
5	**2.1**	~ 500	26.0	4.2	5	Worst-case
6	**2.2**	~ 500	50.0	4.2	5	Best-case
7	**3**	~ 500	74.0	5.3	2	Median
8	**3.1**	~ 500	50.0	5.3	2	Worst-case
9	**3.2**	~ 500	100.0	5.3	2	Best-case
10	**6**	~ 500	187.7	6.1	2	Median
11	**6.1**	~ 500	100.0	6.1	2	Worst-case
12	**9**	~ 500	571.0	7.0	2	Median
13	**9.1**	~ 500	400.0	6.1	2	Worst-case
14	**13**	~ 5,000	38.0	4.2	5	Median
15	**13.2**	~ 5000	50.0	4.2	7	Best-case
16	**16**	~ 5,000	74.0	5.3	7	Median
17	**16.2**	~ 5000	100.0	5.3	7	Best-case
18	**19**	~ 5,000	187.7	6.1	7	Median
19	**19.1**	~ 5,000	100.0	5.3	7	Worst-case
20	**19.2**	~ 5,000	400.0	6.1	7	Best-case
21	**22**	~ 5,000	571.0	7.0	7	Median
22	**22.1**	~ 5,000	400.0	6.1	7	Worst-case
23	**22.2**	~ 5,000	1200.0	7.0	7	Best-case

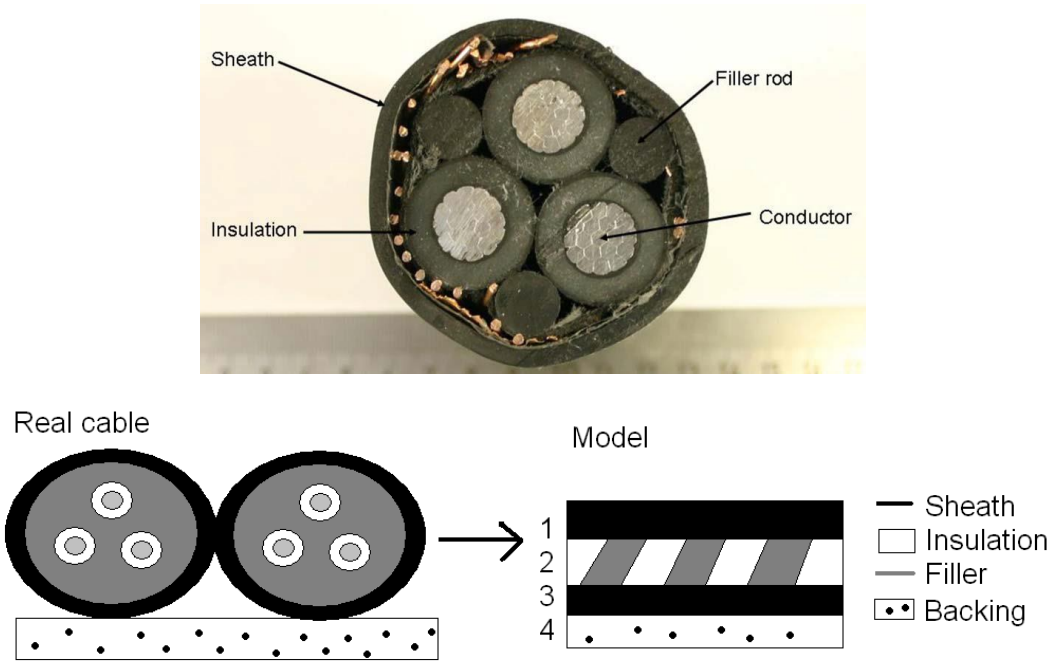
Fire modeling was used to simulate each fire scenario and determine the fire consequences on FSSA cables for the different fire zones. FDS Version 4.05 [21] was used for this study, as NUREG/CR-6850 uses FDS as its main fire-modeling tool [2].

All FDS models were designed to predict the worst-case fire scenarios, and the simulation time was set to 60 min. The firefighting departments of all Canadian nuclear power plants are required to be available to fight all fires within 15 minutes from the time of detection. A 60-minute fire was selected as a worst-case scenario with no manual interventions. Worst-case fire scenarios were obtained by assuming the following:

1) All combustible materials are adjacent and not distributed in each fire zone.
2) All combustible materials in a zone would be involved in every fire.
3) All combustible materials in a zone would undergo total combustion during a fire.
4) Fires will have sufficient air supply for continuous burning (i.e., well-ventilated fuel-controlled fires).
5) No credit is given for the possibility of suppressing the fires.

The miscellaneous power cable components of the fuel load power cable was modelled as a complete cable, neglecting the small amount of additional plastics. The approximation of the cable structure as a planar surface is illustrated in Figure 22.

Figure 22: Real cable versus model [85]



The first and third layers consist of cable sheath material, while the second layer is a mixture of insulation and filler materials. Conductors are not combustible and are therefore neglected in the model for simplicity and to save computational time in large-scale simulations [85, 86]. The fuel load survey performed, Shalabi, H., and Hadjisophocleous, G., [65] identified different diameters for power cables in Canadian NPPs, with the most common and largest diameter of cable being 6 cm, Shalabi, H., and Hadjisophocleous, G., [65]. The FDS input data used for the cables are presented in APPENDIX C.

All walls, ceilings, and floors in the 1,230 fire zones were made of concrete. The mesh sizes used for all simulations were 10 cm x 10 cm x 10 cm, as per NIST validation and verification study [21], in which FDS simulations were performed for similar size fire zones According to that study, FDS can reliably predict gas temperatures, major gas

species concentrations, and zone pressures to within about 15 %, and heat fluxes and surface temperatures to within about 25 % [21, 22].

In the model, a thermocouple and a heat flux sensor were placed at the location of each FSSA cable in order to measure the temperature and the heat flux that the cable would encounter during the fire simulation. The FSSA cables were always found on the upper one-third of a fire zone's wall, Shalabi, H., and Hadjisophocleous, G., [65].

NUREG/CR-6850 [2] defines different values for heat release per unit area (HRRPUA) for power cables as shown in Table 22. To model for the worst-case fire scenario, the highest HRRPUA value (589 kW/m^2) was selected and used in all twenty-three simulations. The FDS input data used in the nine simulations is presented in APPENDIX C.

Table 22. Bench Scale HRRPUA Values [2]

Material	**Bench Scale HRR [KW/m^2]**
XPE/FRXPE	475
XPE/Neoprene	354
XPE/Neoprene	302
XPE/XPE	178
PE/PVC	395
PE/PVC	359
PE/PVC	312
PE/PVC	589
PE, Nylon/PVC, Nylon	231
PE, Nylon/PVC, Nylon	218

7.2 Power Cable of 500 MJ Combustible Load

Fire scenarios 1, 2, 3, 6 and 9 used 500 MJ of power cable combustible load. Since FDS does not accept cylindrical shapes, the 8 m x 6 cm cable was converted to a cubic shape with dimensions 8 m x 6 cm x 6 cm. Power cables are typically laid out in two cable trays with approximately 30 cm between the trays [67], hence, each cable tray has a total volume of length 4 m x width 0.06 m x height 0.06 m ~ 0.0144 m^3. The dimensions of each tray are length 0.33 m x width 0.36 m x height 0.12 m ~ 0.0144 m^3.

Each cable tray Area = 2 (L x W) + 2 (L x H) + 2 (W x H) **(Equation 12)**

Each tray area = 2 (0.33 m x 0.36 m) + 2 (0.33 m x 0.12 m) + 2 (0.36 m x 0.12 m) = 0.4032 m^2; total area = 0.4032 m^2 x 2 = 0.8064 m^2

HRR = HRRPUA x total area **(Equation 13)**

HRR = 589 kW/m^2 x 0.8064 m^2 = 471.2 kW

Assuming a t-square fire growth:

$Q = \alpha t^2$ **(Equation 14)**

Where Q is the HRR (kW), α is the fire growth coefficient (kW/s^2). A value of α= 0.1876 kW/s^2 was chosen representing ultrafast growing fires since the cables have PVC (plastics), and t is time (s).

From Equation 12, 471.2 kW will be attained in 51 seconds.

7.3 Power Cable of 5,000 MJ Combustible Load

Fire scenarios series 13, 16, 19 and 22 used 5,000 MJ of power cable combustible load. To model for the worst-case fire scenario, the highest value of HRRPUA was selected (589 kW/m^2).

With power cables held by a 2-cable tray, each tray has the following dimensions:

Each tray 1.64 m x width 0.36 m x height 0.24 m, each tray area = 2 x (1.64 m x 0.36 m) + 2 (0.36 m x 0.24 m) + 2 (1.64 m x 0.24 m) = 2.12 m^2; total area = x 2 = 4.24 m^2.

HRR = 589 kW/m^2 x 4.24 m^2 = 2,497.4 kW, fire growth coefficient = ultra-fast ~ 0.1876 kW/s^2; Therefore, 2,497.4 kW will be attained in 111 seconds.

There is a cable fire condition for ignition at 205°C that was entered in FDS. This ignition condition will influence the cable fire surface not to all burn at the same time, and instead it will only ignite the cable fire surfaces when it reaches 205ºC. In addition, the HRR ramp was set using the ultra-fast fire growth coefficient, reaching 2,497.4 kW in 111 seconds.

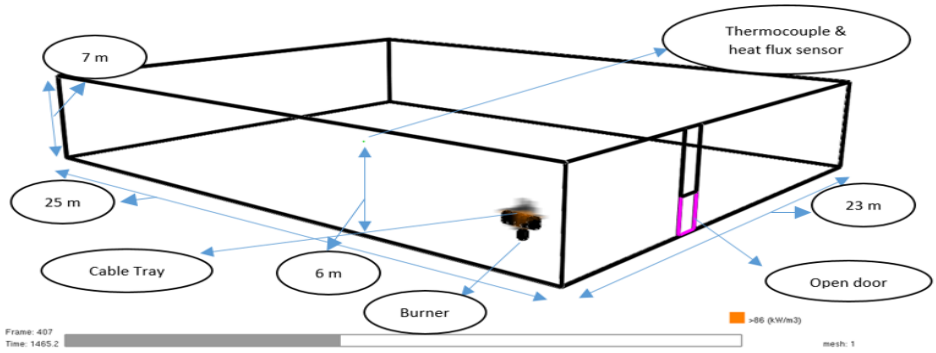
7.4 Results and Discussion

All 23 simulations were carried out and the maximum temperature and heat flux at the FSSA cable were identified and compared to qualified and non-qualified cable temperature and heat flux thresholds. Simulation 22 was selected to illustrate the full details for the inputs and outputs. The dimensions of the FSSA fire zone simulation 22 are 25 m x 23 m, and a height of 7 m. The fire is 7 m away from the thermocouple and heat flux sensor (FSSA cable location) and the height of both the thermocouple and heat flux sensor was 6 meters. The total combustible load is 5,000 MJ for the 2-cable trays, with dimensions of each tray of 1.64 m x 0.36 m x 0.24 m. The burner used to ignite the cable tray was placed under the cable trays and had a HRR of 21 kW [2] with an area of 0.3 m x 0.3 m. Figure 23 shows three FDS Smokeview figures. The first figure shows the fire zone dimensions and the ignition of the cable trays. The figure also shows the distance of 7 m between the cable tray fire and the FSSA cable and the open door in the

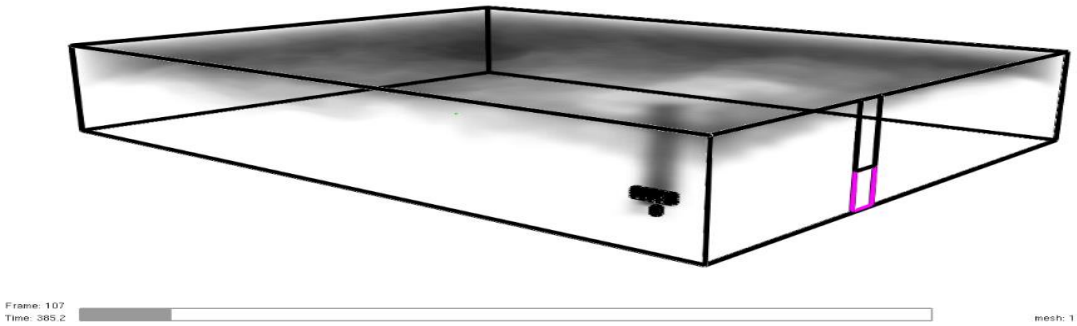
FSSA fire zone. The second figure shows the smoke soot and the spread of smoke right after ignition at around 5 minutes. As shown in the figure, the smoke starts climbing up to the ceiling of the fire zone. The third figure shows the 3-dimensional view of temperature distribution during the fire at around 6 minutes. This is the fire growth period of the fire and temperature and heat flux will continue increasing until the fire is fully developed until it reaches a steady state.

Figure 23. FDS Smokeview illustrations

a)



b)



c)

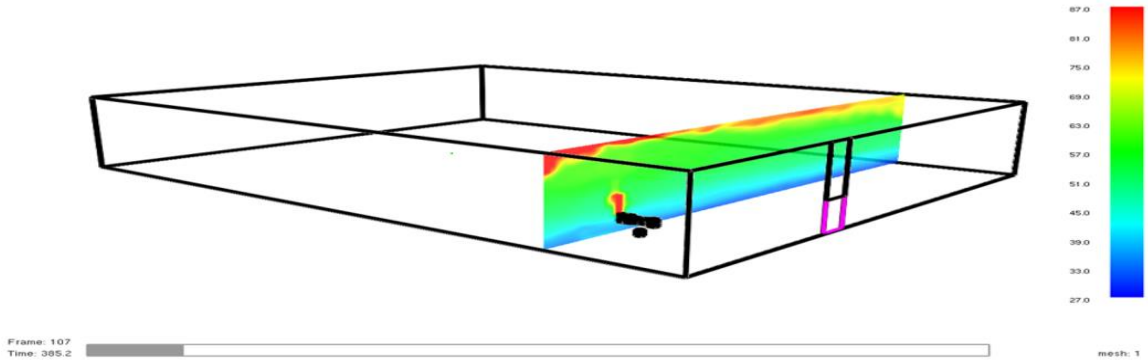


Figure 24 shows the temperature, heat flux and heat release rate (HRR) at the location of the FSSA cable for scenario 22, which has a fire load of 5,000 MJ cable tray fire. As shown in the temperature vs. time curve, the temperature rapidly increases in the first 10 minutes, the rate of increase steadily decreases until 40 minutes, and the temperature decreases after. The figure shows that the temperature increases with time and reaches 161°C at 38 minutes. The curve in Figure 24: Heat Flux vs. Time shows a similar trend for the heat flux near the FSSA cable, which reaches a maximum value of 1.45 kW/m2. The curve in Figure 24: HRR vs. Time shows that the HRR reached a peak value of 1,850.0 kW at 38 minutes and decreased rapidly to less than 600 kW. The increase in the heat release rate at 45 min shown in Figure 24 is a result of the fire spreading to an unburnt part of the cable. This causes a slight increase in temperature and heat flux at around 45 minutes.

Figure 24. Simulation 22 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

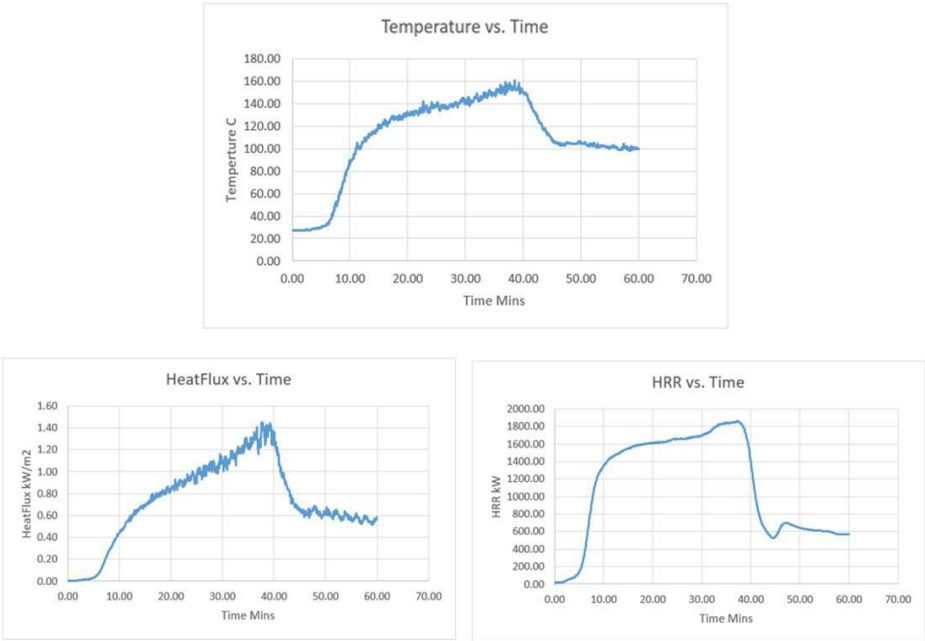


Table 23 shows the maximum temperatures, heat fluxes and HRR values with their corresponding times predicted by FDS for all simulations. All twenty-three simulations followed a similar pattern as discussed for Simulation 22. Table 24 shows the maximum temperatures and heat fluxes for FDS simulations from Table 23, which increased by 25% to account for the FDS error margin.

Table 23. Summary of the results of the twenty-three FDS simulations

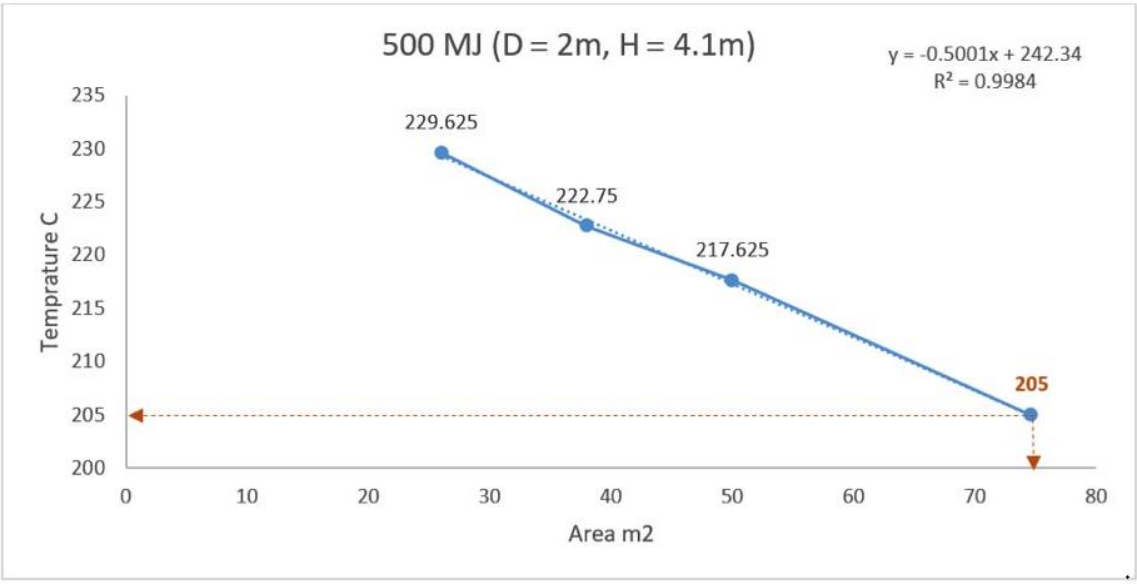
Fire Scenario number	Maximum Temperature (°C)	Corresponding Time (min)	Maximum Heat flux (kW/m$^{2)}$	Corresponding Time (Min)	Maximum HRR (KW)	Corresponding Time (min)
1	178.2	19.5	0.4	19.5	450.0	18.1
1.1	183.7	19.5	0.3	19.5	449.4	18.2
1.2	174.1	19.5	0.3	19.5	450.0	18.1
2	135.2	19.5	0.3	19.5	439.0	17.7
2.1	172.4	19.5	0.3	19.5	439.0	17.7
2.2	104.9	19.5	0.3	19.5	439.0	17.7
3	159.5	18.8	0.3	18.8	594.0	17.8
3.1	178.2	18.8	0.3	18.8	594.0	17.8
6	121.0	16.8	0.3	22.1	537.0	11.3
6.1	127.0	16.8	0.3	22.0	537.0	11.3
9	90.6	29.6	0.2	29.6	571.0	27.6
9.1	94.1	29.5	0.2	29.6	571.0	27.6
13	402.2	35.6	8.1	35.6	2,085.0	11.9
13.2	475.0	35.5	12.2	35.5	2,084.0	11.8
16	363.0	37.0	5.5	37.0	2,556.0	32.7
16.2	342.0	37.0	5.0	37.0	2,550.0	32.6
19	266.2	30.0	1.7	30.0	2,209.0	31.2
19.1	342.0	30.0	5.0	30.0	2,200.0	31.1
19.2	171.0	30.0	1.0	30.0	2,200.0	31.2
22	161.0	38.4	1.45	38.4	1,800.0	35.0
22.1	181.2	38.4	1.0	38.4	1,800.0	35.0
22.2	93.1	38.4	0.2	38.4	1,800.0	35.0

Table 24. Maximum Temperature and Heat Flux results increased by 25%

Simulation Number	Fire Scenario Number	Maximum Temperature (°C)	Maximum Heat Flux (kW/m^2)
1	**1**	223	0.5
2	**1.1**	230	0.4
3	**1.2**	218	0.4
4	**2**	169	0.4
5	**2.1**	215	0.4
6	**2.2**	131	0.4
7	**3**	199	0.4
8	**3.1**	223	0.4
9	**3.2**	159	0.4
10	**6**	151	0.4
11	**6.1**	159	0.4
12	**9**	113	0.3
13	**9.1**	118	0.3
14	**13**	503	10.0
15	**13.2**	594	19
16	**16**	454	9
17	**16.2**	428	6.3
18	**19**	333	3
19	**19.1**	428	6.3
20	**19.2**	241	1.3
21	**22**	201	1.8
22	**22.1**	226	1.3
23	**22.2**	117	0.3

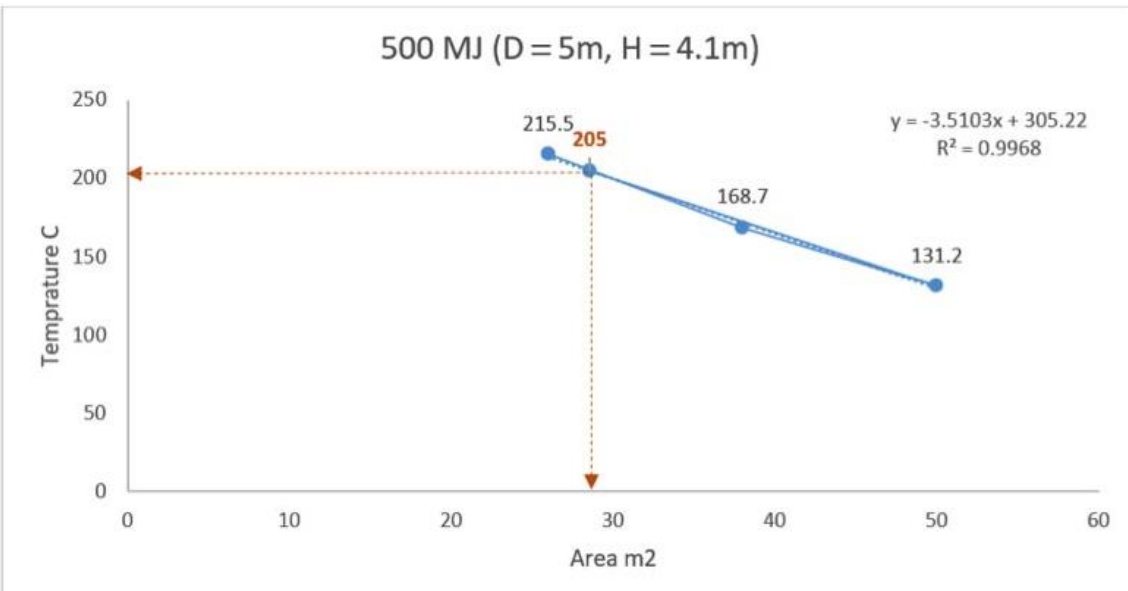
Simulations 1 to 3 were performed to investigate the impact of fire zone area on the temperature and heat flux at the FSSA cable. The simulations were done using three fire zone areas: 26 m^2, 38 m^2 and 50 m^2, height of 4.2 m, a combustible load of 500 MJ and a distance of 2 m from the fire to the FSSA cable. Figure 25 shows the results of the simulations. The figure shows that the FSSA cable temperature decreases as the area of the fire zone increases. For smaller areas, the cable temperature is higher than the threshold value of 205ºC. The temperature reaches 205°C when the area is 74.6 m^2.

Figure 25. Fire Scenario number 1, 1.1 and 1.2 (Simulations 1, 2 and 3)



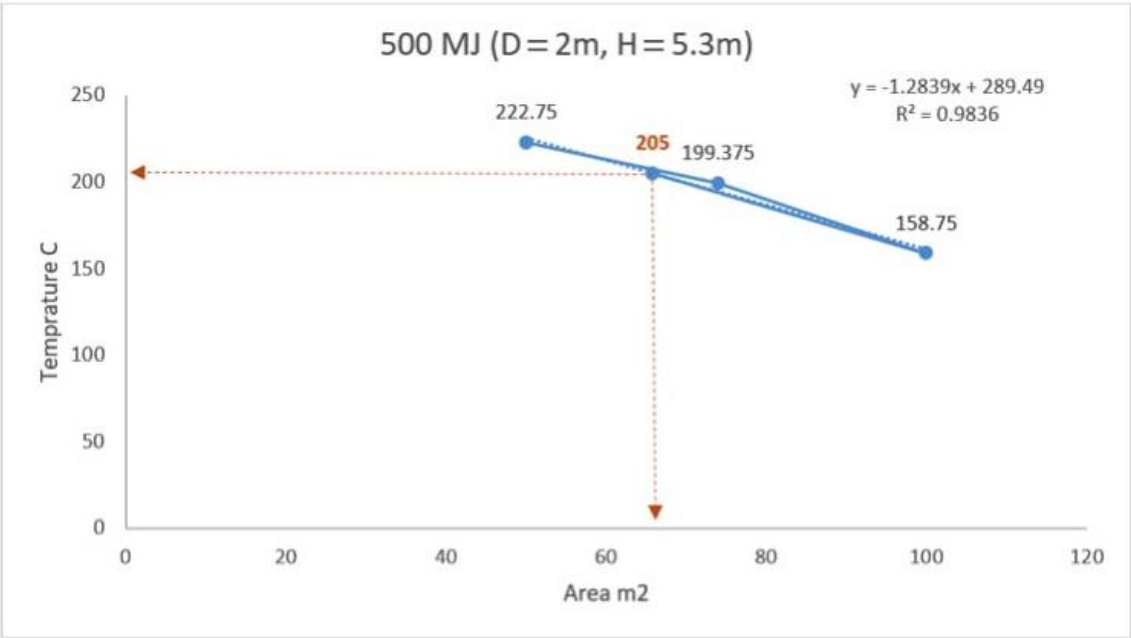
Simulations 4 to 6 were performed to compare the impact of fire zone temperature and heat flux at the FSSA cable to simulations 1 to 3. Simulations 4 to 6 have the same areas, heights and combustible loads used in simulations 1 to 3. However, Simulations 4 to 6 have a distance of 5 m between the fire and the FSSA cable compared to 2 m in simulations 1 to 3. Figure 26 shows that the cable temperatures for simulations 4 to 6 were less than the temperatures for simulations 1 to 3 in Figure 26. This shows that the FSSA cable temperature increases as the distance between the fire and the FSSA cable decreases. Figure 26 shows that the temperature reaches 205°C when the area is 28.5 m^2.

Figure 26. Fire Scenario number 2, 2.1 and 2.2 (Simulations 4, 5 and 6)



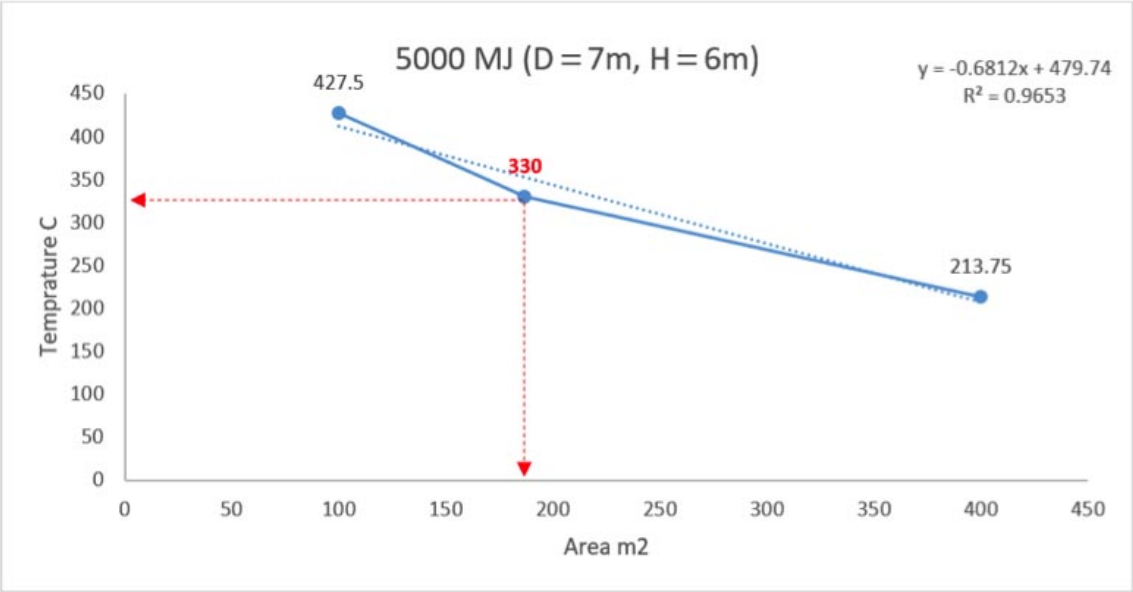
Simulations 7 to 9 were performed to study the impact of the fire zones with areas of 50 m^2, 74 m^2, and 100 m^2 respectively on the temperature and heat flux at the FSSA cable. Figure 27 shows that the FSSA cable temperature decreases as the area of the fire zone increases as observed in the previous simulations. The simulations used 500 MJ combustible load and a height of 5.3 m with a distance of 2 m of the FSSA cable from the fire. Figure 27 shows that the temperature reaches 205°C when the area is 65.8 m^2.

Figure 27. Fire Scenario number 3, 3.1 and 3.2 (Simulations 7, 8 and 9)



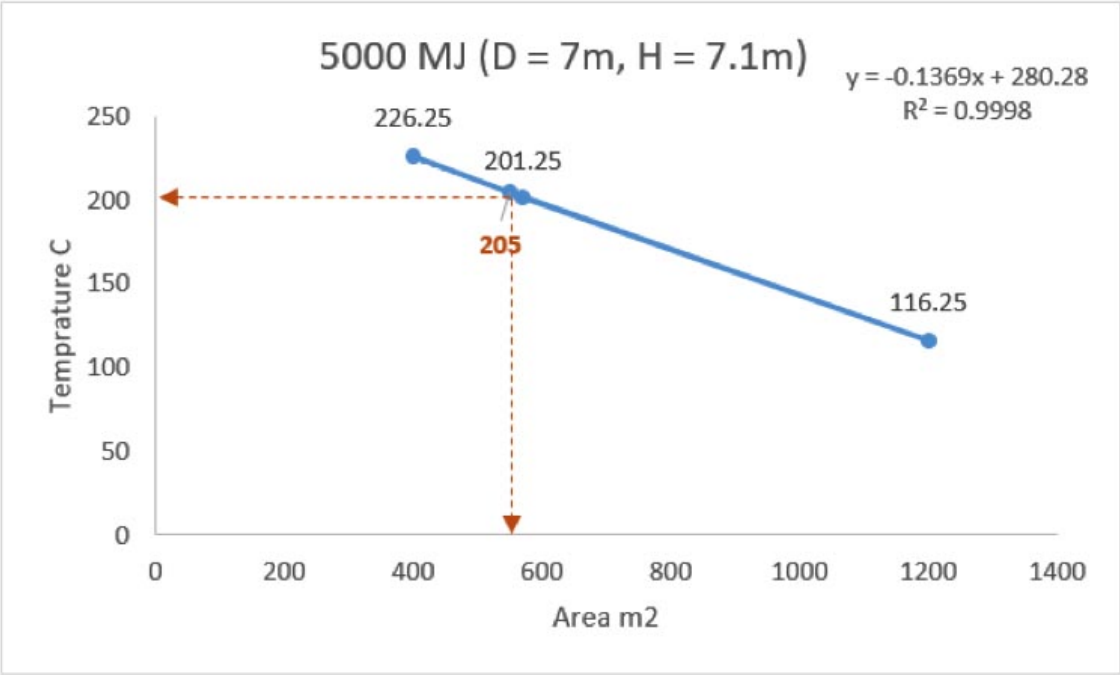
Simulations 18 to 20 were performed to examine the impact of the fire zones areas with a combustible load of 5,000 MJ. Figure 28 shows the same trend again as in the previous simulations. The FSSA cable temperature decreases as the area of the fire zone increases. Figure 28 shows that for the largest area of 400 m^2 the temperature at the FSSA cable is 213ºC, which is higher than the threshold 205°C for non-qualified cables. For smaller areas, the cable temperature is higher than the threshold value of 330°C for qualified cables. Figure 28 shows that the temperature reaches 330°C when the area is 187.7 m^2.

Figure 28. Fire Scenario number 19, 19.1 and 19.2 (Simulations 18, 19 and 20)



Simulations 21 to 23 were performed to investigate the impact of the fire zones areas with a combustible load of 5,000 MJ as in simulations 18 to 20, except using larger areas and heights (400 m^2, 571 m^2 and 1200 m^2 and a height of 7.1 m). Figure 29 shows that for the smallest area of 400 m^2, the temperature at the FSSA cable is 216.25ºC, which is higher than the threshold of 205°C for non-qualified cables and less than the threshold of 330°C for qualified cables. The temperature reaches 205°C when the area is 580 m^2.

Figure 29. Fire Scenario number 22, 22.1 and 22.2 (Simulations 21, 22 and 23)



Simulation 11 has an area of 100 m^2, which represents the worst-case scenario for simulations 10 and 11 for an area range of between 100 m^2 to 400 m^2. Simulation 13 has an area of 400 m^2, which represents the worst-case scenario for simulations 12 and 13, with an area range of 400 m^2 to 1200 m^2. The output temperature and heat flux output for both simulations 11 and 13 did not reach the threshold for both the qualified and non-qualified cables. Therefore, simulations 10 and 11 will result in lower temperature and heat flux outputs and will not damage the qualified or non-qualified FSSA cables.

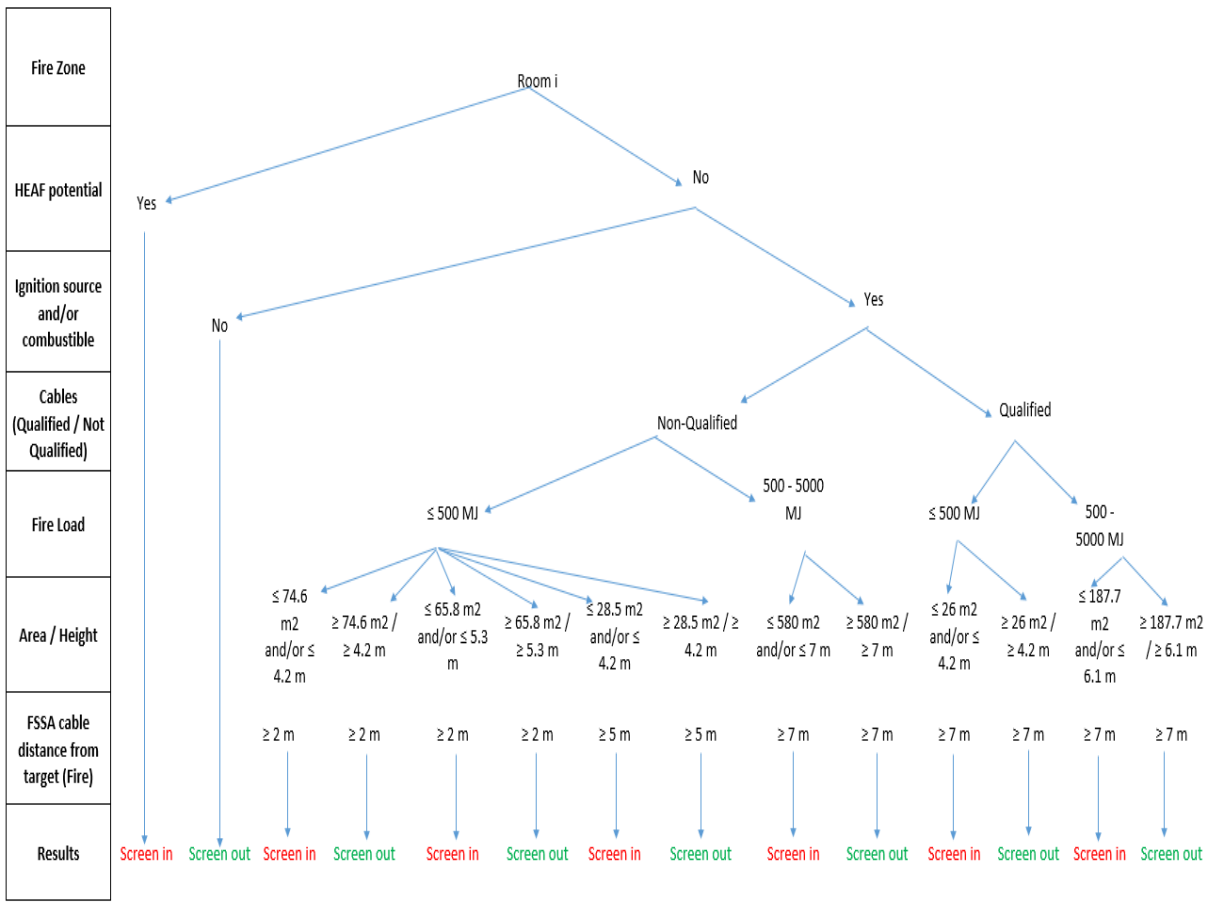
Fire simulations 15 and 17 represents the best-case scenarios for the area range 26 m^2 – 50 m^2 (simulations 14 and 15) and 50 m^2 – 100 m^2 (simulations 16 and 17) respectively. Simulation 15 has an area of 50 m^2, and simulation 17 has an area of 100 m^2. Both simulations 15 and 17 temperature and heat flux outputs exceeded the threshold for both the qualified and non-qualified cables. Therefore, simulations 14 and 16 will result in higher temperature and heat flux outputs and will damage the FSSA cables.

Qualitative Screening Decision Tree Step

The results of the twenty-three FDS simulations were used to develop a decision tree shown in Figure 30, which can be used to screen in and out fire zones. According to the tree, any fire zone that has HEAF potential must be screened in and quantitatively analyzed, any fire zone that does not have an ignition source and/or combustible can be screened out from further analysis. The remaining zones are screened in or out depending on the expected impact of the fires in these zones. All other remaining fire zones that are not included in the screening tree will have to be screened in for further quantitative analysis.

For example, in a given fire zone that has an ignition source, the cables are non-qualified, and the combustible load is < 500 MJ: any area under 74.6 m^2 and a height of less than 4.2 m with a distance of FSSA cable and fire of 2 m must be screened in for further analysis. On the other hand, if the area is greater than 74.6 m^2 and the height is equal or greater than 4.2 m with a distance of FSSA cable and fire of 2 m or more, the fire zone can be screened out from further analysis.

Figure 30. Qualitative Screening Decision Tree [76]



Chapter 8: Case Studies using CANDU Fire PRA Model, Shalabi, H., and Hadjisophocleous, G., [87]

In this chapter, two FSSA fire zones in Canadian nuclear facilities were selected to demonstrate the use of CANDU Fire PRA model and the proposed qualitative screening step. The PRA was carried out using Canadian data from the CANDU fuel surveys. The first fire zone selected is a monitoring room and the second selected fire zone is a process room. The fire zones were selected because of their significance in any nuclear plant. As seen in Chapter 3, the process room has the highest percentage for fires starting in CANDU plants, while the selection of the monitoring room was based on its importance for the safety of the plant.

8.1 Monitoring Room Description and Analysis

The monitoring room was selected from the CANDU fire load survey, Shalabi, H., and Hadjisophocleous, G., [65] and had the following dimensions: 6 m x 14.5 m x 4.5 m high. The combustible load was 478.8 MJ, consisting mainly of electric cables. The FSSA cable was 5 m away from the combustibles and at a height of about 4 m. All ceilings, walls, and floors were made of concrete. The monitoring room had no sprinklers and did not have any detection equipment. It was assumed that the cause of ignition in the room was due to electrical fault.

8.2 Applying CANDU Fire PRA to the Monitoring Room

Applying the first three tasks of the CANDU Fire PRA to the monitoring room gave the following:

Task 1: Plant Boundary Definition and Partitioning

The monitoring room had concrete walls/ceilings/floors that were considered fire barriers. They all had a fire resistance rating of more than one hour. All elements of the barriers, such as fire doors and penetrations, exceeded the minimum rating of one hour.

Task 2: Fire PRA Components and Equipment Including Cables

The monitoring room contained non-qualified FSSA cables, and therefore was identified as an FSSA cable credited fire zone that could affect safety systems during a fire event.

Task 3: Qualitative Screening

The monitoring room has a combustible load of 478.8 MJ with an area of 87 m^2, a height of 4.5 m and a distance between FSSA cable and the nearest combustibles of about 5 m. Using the decision tree in Chapter 7, the combustible load < 500 MJ, the area is > 74.6 m^2. Therefore, the fire zone can be screened out from further analysis.

8.3 Process Room Description and Analysis

A large size process room was selected from the CANDU fire load survey, Shalabi, H., and Hadjisophocleous, G., [65] and had dimensions of 25 m x 23 m x 7 m = 4,025 m^3. The combustible load was roughly 4,400 MJ, consisting mainly of cable trays. The FSSA cable is less than 7 m away from the combustibles and at a height of 6 meters. The FSSA cable was unqualified cable. All ceilings, walls, and floors were made of concrete. The process room had 16 sprinkler heads and 2 heat detectors. It was assumed that the cause of ignition in the room was due to electrical fault.

The damage threshold temperature for IEEE-383 qualified cables [88] is 330°C and for non-qualified cables is 205ºC. The damage threshold heat flux for IEEE-383 qualified cables [88] is 11 kW/m^2 and for non-qualified cables is 6 kW/m^2.

8.4 Applying CANDU Fire PRA to the Process Room

Applying the tasks of the CANDU Fire PRA to the monitoring room gave the following:

Task 1: Plant Boundary Definition and Partitioning

The process room's concrete walls/ceilings/floors were considered fire barriers, which had a fire resistance rating of more than one hour. All elements of the barriers, such as fire doors and penetrations, exceeded the minimum rating of one hour.

Task 2: Fire PRA Components and Equipment Including Cables

An FSSA cable inside the fire zone (process room) was identified as an FSSA component in this analysis that could affect safety systems during a fire event.

Task 3: Qualitative Screening

The monitoring room had a combustible load of roughly 4,400 MJ, an area of 575 m^2, a height of 7 m and the distance between the fire and the FSSA cable was 7 m. The FSSA cable is an unqualified cable. Using the decision tree in Chapter 7, the combustible load is between 500 – 5,000 MJ, and the area is > 580 m^2, and the distance between the combustibles and fire is 7 m. Therefore, the fire zone is screened in for further analysis.

Task 4: Ignition Frequency

Using the two-stage Bayesian model as shown in Chapter 3 of this thesis, the probability of a fire to occur in a CANDU reactor in Canada is 1.19 E-01 per reactor year.

Task 5: Circuit Failure Analysis

For this fire scenario, Circuit Failure Analysis is not in the scope of this analysis.

Task 6: Fire Modelling

The FSSA cable in the process room had an impact on the reactor's safety systems and on the operation of the components. For this step, two FDS simulations are conducted. The first FDS model had sprinkler and detection systems, and the second simulation did not include those systems. It was assumed that fire consumes all combustibles. The outcome of these two FDS simulations is essential to determine the consequences required for the event tree analysis. Both FDS models assumed to have a power cable tray fire due to an electric fault. The fire occurred around midnight when the room was unoccupied. There were no windows or openings in this room except for the fire door. For maximum ventilation, the door was propped open in both simulations.

Concrete properties [2]:

- Thermal Conductivity = 0.001 kW/mK
- Density= 2000 kg/m^3
- Specific heat 0.88 kJ/kg
- Wall thickness 0.3 m

Cable tray (Nylon/PVC) properties [2]:

- HRRPUA for cables = 231 kW/m^2

Cable trays:

There were two cable trays, with each tray's LWH being 2 m x 1 m x 0.3 m = 0.6 m^3.

From Equation 11, One cable tray area = 2 (L x W) + 2 (L x H) + 2 (W x H)

Each cable tray area = 2 (2 m x 1 m) + 2 (2 m x 0.3 m) + 2 (1 m x 0.3 m) = 5.8 m^2

Total area = 5.8 m^2 x 2 = 11.6 m^2

From Equation 13, the t-squared parabolic growth equation is given by $Q = \alpha t^2$

Where Q is the HRR (kW), α is the fire growth coefficient (kW/s^2), and t is time (s).

From Equation 12, HRR = HRRPUA x total area

HRR = 231 kW/m^2 x 11.6 m^2 = 2679.6 kW, α = ultra-fast ~ 0.1876 kW/s^2; Therefore, 2,679.6 kW will be attained in 120 seconds.

There is a cable fire condition for ignition at 205°C that was entered in FDS. This cable surface will only ignite when it reaches 205ºC. In addition, the HRR ramp was set using the ultra-fast fire growth coefficient, reaching 2,679.6 kW in 120 seconds.

Total MJ = HRRPUA X AREA X time X F (area under HRR vs time curve)

Total MJ = {231 kW/m^2 x 11.6 m^2 x (1635 s -127 s) x 1} + {(2) 231 kW/m^2 x 11.6 m^2 x 127 s x 0.5} = 4,040,836.8 MJ + 340,309.2 MJ = 4,381.2 MJ

FDS HRR input:

```
&RAMP ID='Cable_RAMP_Q', T=0.0, F=0.0/
&RAMP ID='Cable_RAMP_Q', T=120.0, F=1.0/
&RAMP ID='Cable_RAMP_Q', T=1628.0, F=1.0/
&RAMP ID='Cable_RAMP_Q', T=1755.0, F=0.0/
```

- Simulation conditions:
  - Domain, room dimensions XYZ 25.0 m x 23.0 m x 7.0 m
  - Thermocouple & heat flux sensor XYZ 0.6, 20.3, 6.0
  - Location of cable (First Tray) XYZ 5.0, 14.8, 0.3 to 7.0, 15.8, 0.6
    (Second Tray) XYZ 5.0, 14.8, 0.9 to 7.0, 15.8, 1.2
  - Distance between thermocouple/heat flux sensor and fire is:
    $\{(5.0\ m - 0.6\ m)^2 + (14.8\ m - 20.3\ m)^2\}^{1/2} = 7.0\ m$
  - Door open, vent XYZ 5.0, 13.8, 0.1 to XYZ 6.7, 13.8, 2.6
  - Mesh size # of elements 250 x 230 x 70 = 4,025,000
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m

- Boundary conditions:
  - Burner: HRRPUA 233.3 kW/m^2
  - Burner Area: 0.3 m x 0.3 m
  - Burner Location XYZ 6.0, 14.4, 0.1 to 6.3, 14.7, 0.1

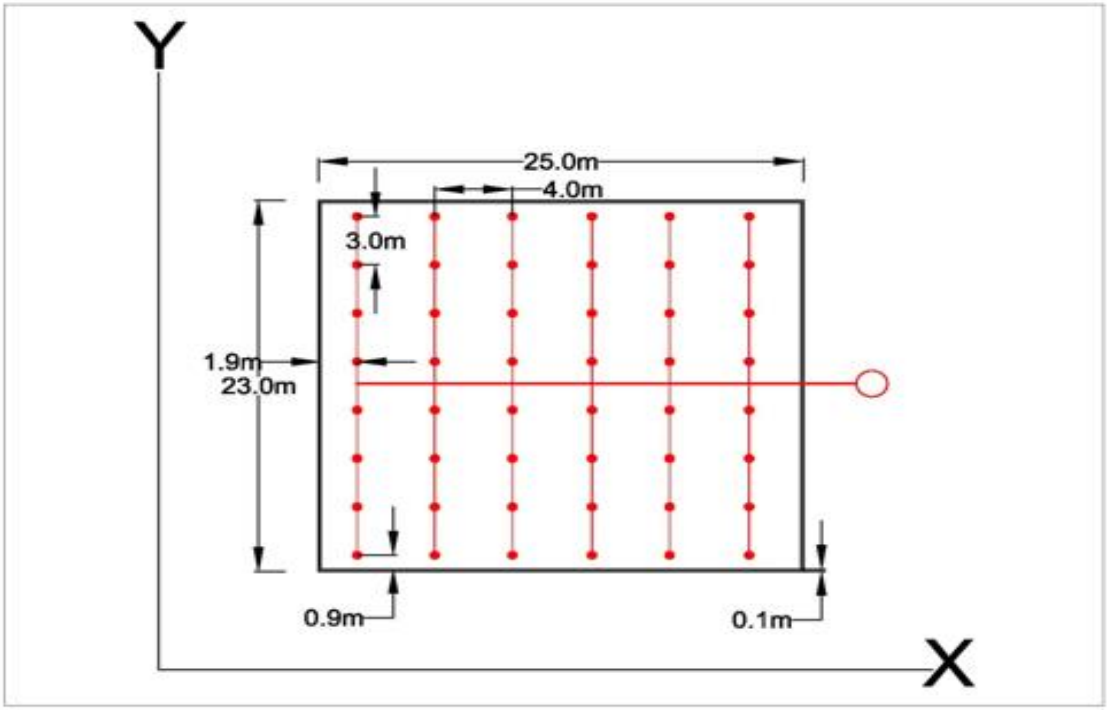
- Sprinkler system conditions:
    - Sprinkler system (wet pipe) has 48 sprinkler heads, the activation temperature =73.0ºC, flow rate is 19.5 gpm, area of operation is 139 m^2, and the spray density is 0.15 gpm/ft^2 or 6.1 mm/min as per NFPA 13 [89]

FDS input: (Simulation 1):

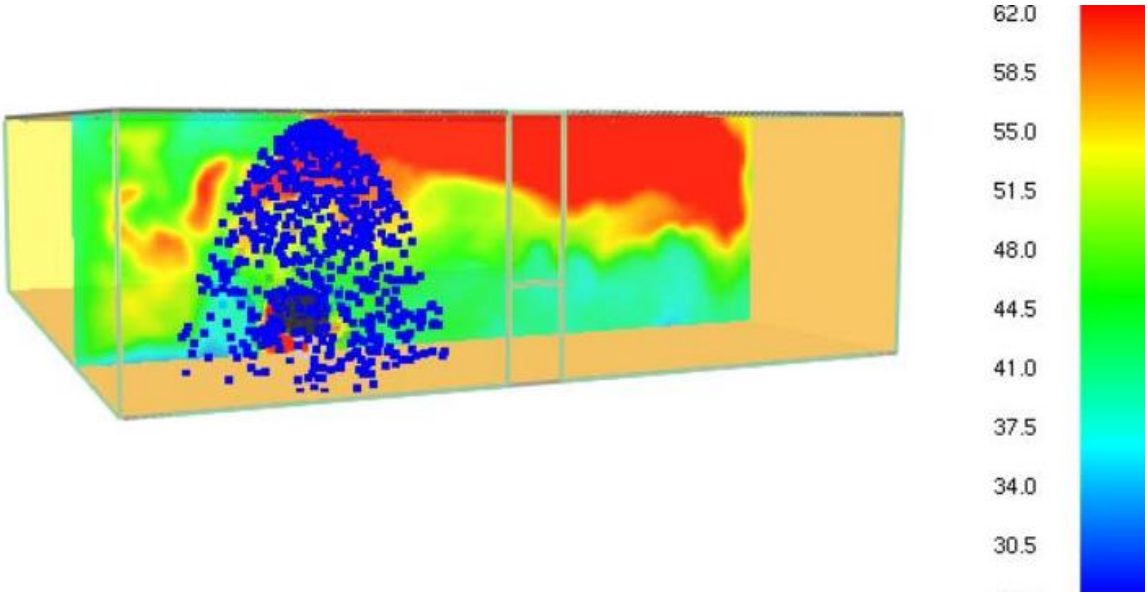
The process room was identified as Ordinary Hazard (Group I) as per NFPA 13 [89]. As per Figure 11.2.3.1.5 in NFPA 13, the spray density for Ordinary Hazard (Group I) is 0.15 gpm/ft^2 or 6.1 mm/min, flow rate for one sprinkler is 19.5 gpm and the area of operation is 139 m^2. There are 48 sprinkler heads in the fire zone of any area of 25 m x 23 m. Figure 31 a) illustrates the spacing of all sprinkler heads in the process room. Figure 31 b) and c) show that the sprinkler spray water does not effectively reach the lower electric tray and the burner. From FDS user's guide [21]: "There are some applications, like the suppression of racked storage commodity fires, where it is useful to allow water droplets to move horizontally along the underside of a solid object. It is difficult to model this phenomenon precisely because it involves surface tension, surface porosity and absorption, and complicated geometry. However, a way to capture some of the effect is to set ALLOW_UNDERSIDE_PARTICLES=.TRUE. on the MISC line.". Therefore, this line of code was added to the FDS model to realistically model the fire under the sprinkler system.

Figure 31. Sprinkler heads spacing in the fire zone

a)



b)



c)

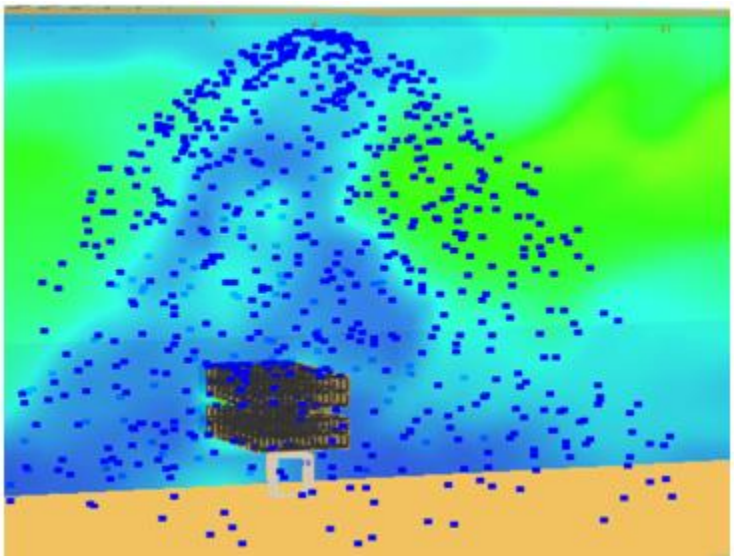
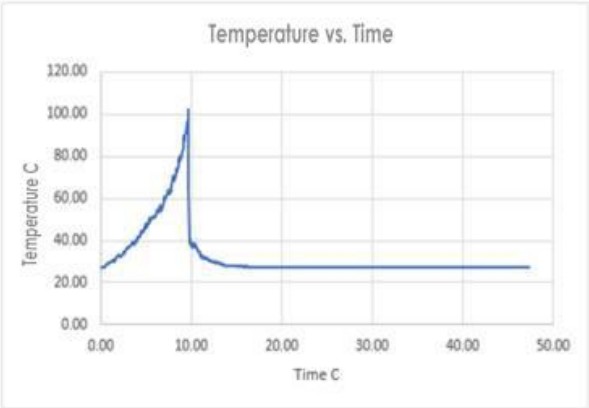
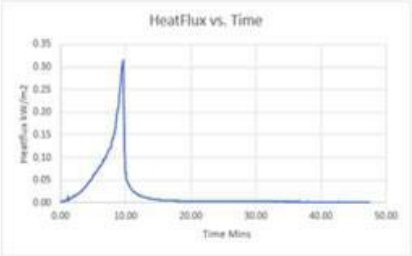
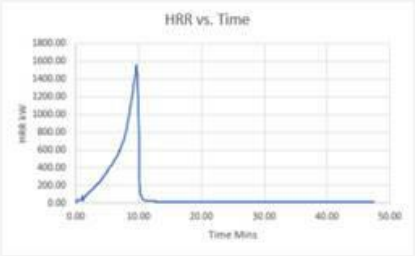


Figure 32. Simulation 1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time



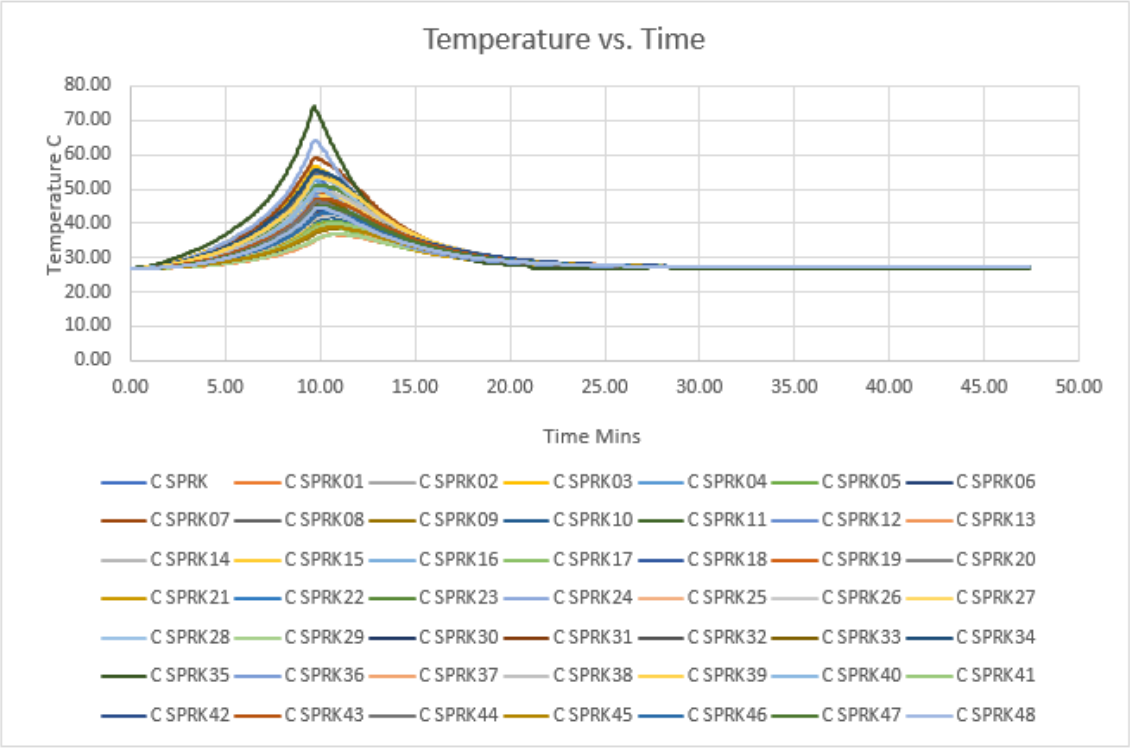




In this simulation, sprinkler SPRK35 activated at around 10 minutes and quickly extinguisher the fire. Figure 32 shows that at the time of sprinkler operation, the maximum temperature reached beside the cables was 102.27°C and the maximum heat

flux of 0.32 kW/m^2 was reached. While, the maximum HRR reached was 1,564 kW. The rest of the sprinkler heads did not activate, as they never reached 73°C. These results are shown in Figure 33.

Figure 33. Simulation 1 Sprinkler system: Time vs. Temperature



Simulation 2 Output

Figure 34 shows three different graphs for temperature, heat flux and HRR outputs. The figure also illustrates that maximum temperature at the thermocouple (FSSA cable), which was 7 m away from the fire, was 245.7°C and was reached at 29.5 minutes from the start of the simulation. Figure 34 also shows that the maximum heat flux measured by the heat flux sensor (FSSA cable), which was 7 m away from the fire, was 2.0 kW/m^2 and was reached at 29.5 minutes, and the maximum HRR reached is 2,930 kW at 24.3 minutes.

Figure 34. Simulation 2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

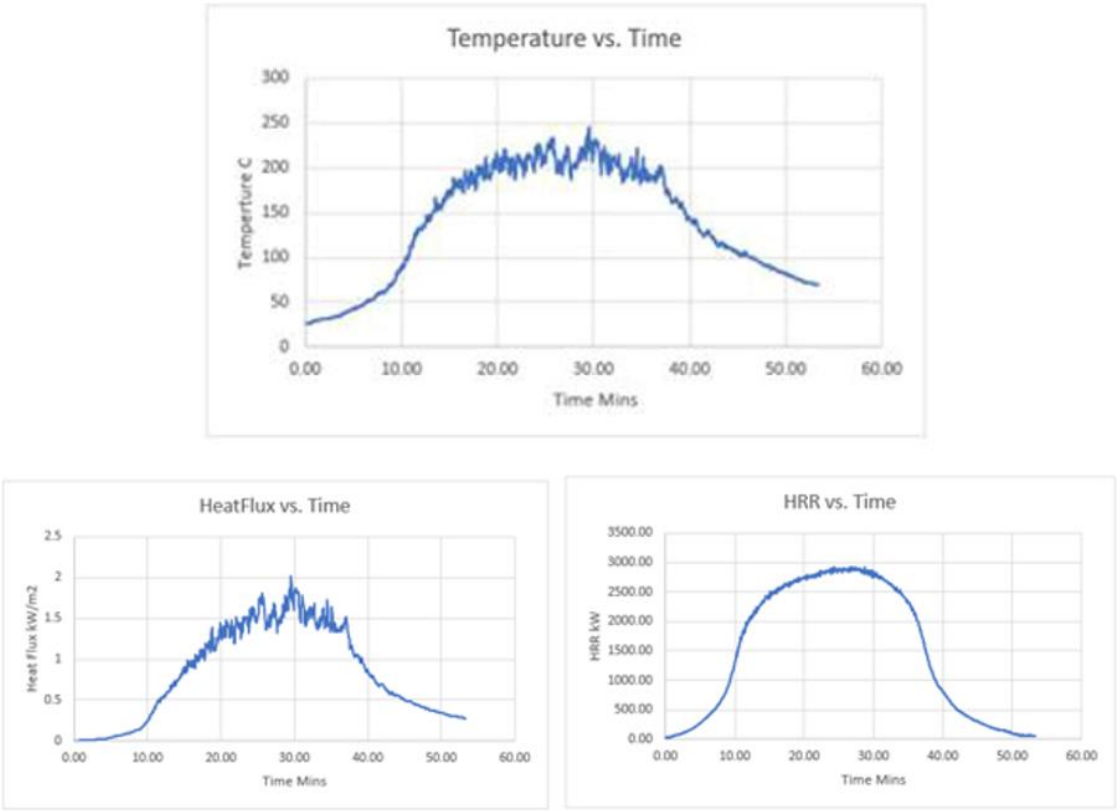


Figure 32 shows that the fire was extinguished after the activation of the sprinkler system after 10 minutes. While Figure 34 shows the same fire with no automatic or manual interventions. The results of these two figures are used to construct the event tree shown later in this chapter.

Task 7: Post-Fire Human Reliability Analysis (HRA)

For this fire scenario, HRA in Fire PRA is not in the scope of this analysis.

Task 8: Quantification of Fire Risk

Event tree analysis (ETA) provides a process that combines the likelihoods and fire protection system(s) (FPS) success probability and consequences to determine the risk of the fire. The ETA steps are divided into:

1. Initiating fire source events

2. Fire protection system(s) performance
3. Case study results
4. Consequences at the target

Risk = initiating event likelihood x FPS success probability x consequences

1. Initiating Fire Source Events

As shown in Chapter 3 of this thesis, Equation 9 was used to calculate the fire occurrence in CANDU reactor in Canada to be 1.19 E-01 per reactor per year.

2. Fire Protection System Performance

The failure rate of a fixed wet-pipe sprinkler suppression system is 0.05 % [90]. Heat detectors had a failure rate of 0.32 % [91], which includes all heat detectors. The on-site manual fire department is credited for fire suppression in virtually all areas of the plant. A review of plant training records and past fire drills was found appropriate to determine an anticipated response time of less than 6 - 8 minutes for a process room. All past fire drills and actual fires on this site were 100 % successfully extinguished. For the purpose of this analysis, it was assumed that the success rate would be 99.9 %.

3. Case Study Results

As illustrated in the fire scenarios developed in this chapter, there were two FDS simulations. Both simulations 1 and 2 had the same area/height and combustible loads. Simulation 1 had two heat detectors and a sprinkler system. The maximum temperature and heat flux output in Simulation 1 did not exceed the threshold for the unqualified FSSA cable (205°C and/or 6 kW/m^2). While for Simulation 2, there was no fire protection system in place and the output temperature exceeded the threshold of 205ºC, which could damage the FSSA cable. Table 25 shows the results of the simulations.

Table 25. Simulation results

Simulation number	Maximum Temperature (ºC)	Maximum Heat Flux (kW/m^2)
Simulation 1	102.27	0.35
Simulation 2	245.7	2.00

4. Consequences at the Target

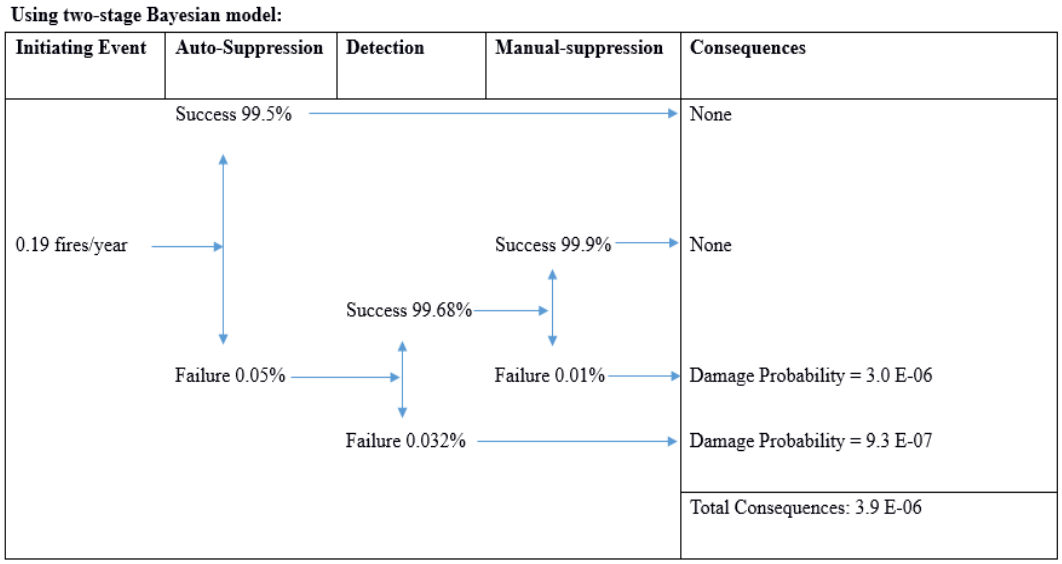
Simulation 1's heat detection system reached its activation temperature (73ºC) after 11 minutes at the start of the simulation. From past fire drills in this nuclear plant, it was found that the anticipated response time was between 6-8 minutes for a process room. Therefore, the fire response will have sufficient time to put out the fire.

Fire Protection System (FPS) Controls:

1) Sprinkler system success rate of 99.5 % [90]
2) Detection system success rate of 99.68 % [91]
3) Fire department success rate of 99.9 % (plant-specific - internal communication and with the licensee and CNSC.)

The fire event outcome tree is shown in Figure 35. The total FSSA cable damage probability for the process room is 3.9 E-06 using the two-stage Bayesian model. If the FSSA-unqualified cable was switched to a qualified FSSA cable, this will exclude the process room from any FSSA cable damage probability. The Canadian regulatory limits for core damage frequency is defined in REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants" [92] to be 10 E-04 per reactor-year for existing plants.

Figure 35. Case Study event-tree: (using two-stage Bayesian model)



8.5 Summary and conclusion

In this chapter, the CANDU Fire PRA was applied to two FSSA fire zones. In order to construct the event tree, two FDS simulations were carried out. The first simulation used a sprinkler system to extinguish the fire, while the second simulation was done without the sprinkler system. Both simulations are identical in terms of combustible load, area, height and distance between combustibles and FSSA cable. The sprinkler system success rate and the detection system success rate from literature were used to complete the event tree in Step 8 for the quantification of the fire risk. A plant specific fire department success rate was also added to finalize Step 8 in the Fire PRA. Using the fire frequency from the two-stage Bayesian discussed in Chapter 3 of this thesis, the total fire risk was determined to be 3.9 E-06.

Chapter 9: Conclusions and Recommendations

9.1. Summary

The objective of this research was to develop a CANDU Fire PRA methodology. A series of tasks were defined to create this CANDU Fire PRA methodology. The first task was to analyze the CANDU Fire Database that maintains all the CANDU fires, followed by the development of a Fire PRA methodology for CANDU reactors. One of the most important steps in the CANDU Fire PRA is the qualitative screening step. In order to develop a qualitative screening step (decision tree) for CANDU reactors in Canada, a fuel survey for all FSSA rooms in CANDU reactors in Canada was carried out to define the critical fire scenarios and the typical combustible loads. The CANDU Fire PRA model was then implemented on two existing FSSA rooms in Canada. In addition, there was a section introduced for HEAF fire events in CANDU reactors and recommendations made to mitigate its occurrence and consequences.

An analysis of the CANDU fire database was carried out. The CANDU fire database contained 76 + 1 reported fires representative for 552.14 reactor years, 64 + 1 of these fires were reported for CANDU reactor fires (representing an observation period of 547.90 reactor years), and 12 fires were reported from the Gentilly 1 site (representing 4.24 years of observation), which had prototype CANDU-BWR reactors that occurred between the years 1981 and 2017. There was another fire in 2018 to make the total reported fires 77 for the period of 1981 to end of 2019.

There was an increase in reported fires from 1999 to 2005. From 1981 to 1996, the fire reporting criteria for each CANDU licensee were listed in their licence condition handbooks, and from 1996 to 2003, the fire reporting criteria followed “R-99: Reporting

Requirements for Operating Nuclear Power Facilities". From 2003 to 2014, the fire reporting criteria followed "S-99: Reporting Requirements for Operating Nuclear Power Plants", and from 2014 to today, the fire reporting criteria have followed "REGDOC-3.1.1.: Reporting Requirements for Nuclear Power Plants (NPPs)". There is a clear relationship between the increased numbers of fires reported since the transition from listing the fire reporting criteria in regulatory documents from the licence condition handbook in 1996. The first CSA-N293 standard, "Fire Protection for CANDU NPPs", was developed and approved for publication in February of 1997, and the CSA-N293 standard was then adopted in the Canadian NPPs' license condition handbooks. CSA-N293 provides the minimum fire protection requirements for the design, construction, commissioning, operation, and decommissioning of CANDU NPPs, including structures, systems, and components (SSCs) that directly support the plant and protected areas. CSA-N293 also states the fire protection requirements for Canadian NPPs. Some of the CSA-N293 standard requirements include having fire protection programs and performing fire protection assessments (code compliance reviews, fire hazard analyses, and FSSA). CSA-N293 also defines the design and installation requirements of fire protection systems. CSA-N293 implementation requires a lot of effort, personnel, and expertise, and requires time to adapt and mature. This could be an explanation for the decrease in the number of fires after 2005.

A CANDU Fire PRA was developed with nine steps and two optional steps. This CANDU Fire PRA addressed some gaps in NUREG/CR-6850. There are numerous differences between the CANDU Fire PRA and NUREG/CR-6850. Some of these differences are:

- While, NUREG/CR-6850 has been developed to be applied as a plant specific and not a generic tool. One of the main advantages of the CANDU Fire PRA is the qualitative decision tree, as it is an easier and simpler approach for screening by reducing the number of fire zones to be investigated in more detail and scenarios to be analyzed significantly already without needing plant/unit specific details,
- NUREG/CR-6850 has 16 steps, while the CANDU Fire PRA has only nine steps with two optional steps. Due to the higher safety systems of CANDUs compared to LWRs, and the limited number of the Canadian nuclear plants in comparison to the U.S., the CANDU Fire PRA was able to be compressed to nine steps instead of 16,
- The use of Canadian ignition frequencies including both locations and equipment in the CANDU Fire PRA, was compared to the U.S. experience,
- Adding "any emergency power supply" to the FSSA cables in Task 2 of the CANDU Fire PRA. The decision to add and identify "any emergency power supply" to the list was based on the Fukushima accident lessons learned by the IAEA,
- Avoiding the use of HEAF equations from NUREG/CR-6850 and using the recommendations in this research instead,
- Adding the optional steps "Other Combination Hazard(s)-Fire Events" and "Severe Accident Management" are add-ons to the CANDU Fire PRA, as they are based on lessons learned from the Fukushima accident and the severe accident guidelines from the IAEA, and the last and the major difference is the qualitative analysis step, and

- A complete uncertainty and sensitivity analysis for a CANDU Fire PSA for any reference plant is to be performed providing not only mean values for the fire induced core damage frequency but also for quantifying the major uncertainties. This uncertainty and sensitivity analysis will reduce the uncertainties in the CANDU Fire PSA and increase the level of confidence in the fire induced CDF values.

The fuel load survey for all FSSA rooms in CANDU reactors in Canada was done only for fire zones that contain FSSA equipment. CSA N293-12 defines FSSA as an evaluator of the capability to safely shut down and maintain a reactor in the shutdown state with respect to postulated fire damage. There are 19 operating CANDU reactors in Canada at five sites (Bruce A, Bruce B, Darlington, Pickering, and Point Lepreau). Fire load density surveys were carried out for all FSSA fire zones at all 5 sites, and the maximum and average fuel loads for all fire zones are presented. The fire load density surveys included floor areas/ceiling heights, ceiling/wall/floor construction, available suppression and/or detection systems, accessible fire hoses and/or portable extinguishers, available ventilation and/or penetrations, HEAF, all potential ignition sources, and the types and quantities of combustibles. The fuel survey carried out included 1,230 fire zones. A general fire zone list for all sites was developed to combine fire zones with similar functions, and hence 38 general fire zones were developed.

A decision tree was developed for the qualitative analysis step of the CANDU Fire PRA. The decision tree is based on the CANDU FSSA room specifications and required a large number of fire simulations. FDS was used to define the consequences of all fire scenarios. The results of the simulations were used to develop a decision tree as

shown in this thesis, which can assess FSSA fire zones. According to the decision tree, any fire zone that has HEAF potential must be screened in and quantitatively analyzed, and any fire zone that does not have an ignition source and/or combustible can be screened out. The remaining fire zones are assessed based on the expected impact of fires.

Two fire zones were selected as case studies to implement the CANDU Fire PRA to demonstrate the application of the CANDU Fire PRA model in Canadian nuclear facilities and determine the FSSA cable damage probability. The monitoring room was the first selected fire zone and was screened out from further analysis by using the qualitative analysis decision tree. The process room was the second selected fire zone, and two FDS simulations were conducted. The first FDS model had sprinkler and detection systems, and the second simulation did not include any fire protection systems. The outcome of these two FDS simulations was essential to determine the consequences for the event tree analysis. The first simulation's heat detection system reached the activation temperature (73ºC) at around 8.5 minutes of the start of the simulation, and the second simulation showed that the damage temperature of 205°C ± 25 % was reached. From past fire drills at this selected NPP, it was found that the anticipated response time was between 6-8 minutes for a process room; therefore, the fire response would have sufficient time to extinguish the fire.

Two past HEAF fire events in Canada were reviewed in detail. The current HEAF prediction models' issues and limitations were also examined. Two sets of recommendations were produced in this research, with the first set for Canadian nuclear power plants and the second for the international nuclear industry.

9.2 Main Conclusions

An analysis of the CANDU Fire database was done, and from this analysis, the following were concluded:

- A list of initiating rooms, buildings and causes of fire events, detection system, fire extinguishing and manual firefighting performance was developed.

- The probability of a fire was calculated using two-stage Bayesian model as shown in Chapter 3 of this thesis. The probability of a fire to occur in a CANDU reactor in Canada is 1.19 E-01 per reactor year.

A CANDU Fire PRA was developed with nine steps and two optional steps. A fuel load survey was done, and the following were concluded:

- A general fire zone list for all sites was developed to combine fire zones with similar functions, and hence 38 general fire zones were developed,
- Typical combustibles in these FSSA fire zones were defined. The fire zones were classified into three groups: ≥ 500 MJ, 500 MJ - 5,000 MJ and > 5,000 MJ,
- CANDU's FSSA Fire Zone Areas and Heights were categorized into five groups: 26 – 50 m^2; 50 – 100 m^2; 100 – 400 m^2, 400 - 1,200 m^2 and over 1,200 m^2,
- The typical distance between the combustibles and the FSSA cable was found to be 2 m, 5 meters and 7 m,
- The average fuel density for all 1,230 fire zones is 170.1 MJ/m^2,
- The suppression and detection equipment, fire hose and portable extinguisher inside the fire zone availability for all 38 general fire zones were identified,
- The HEAF risk was found in 254 out of the 1,230 general fire zones, and

- Electric faults were the greatest percentage of ignition sources and accounted for 1,049 out of all 1,230 fire zones, at roughly 85 % of all ignition sources.

A qualitative analysis decision tree was created using the FDS output for selected fire zones. Twenty-three fire scenarios were identified and simulated. The temperature and heat flux output from these simulations were used to determine the threshold areas for both qualified and non-qualified cables for different combustibles loads.

Two FSSA fire zones in Canadian nuclear facilities were selected to implement the CANDU Fire PRA model on. The CANDU Fire PRA model determined the FSSA cable damage probability.

9.3 Recommendations for Future Research

There are several areas for research that would advance the CANDU Fire PRA. The first area of research involves developing a new HEAF prediction model, as there is an international need to create a more accurate model. A HEAF correlation is needed to develop a calculation tool to estimate different types of blast effects. This correlation can be verified and validated to OECD HEAF experiments data. The second area of research involves the CANDU Fire PRA's optional tasks: "Other Combination Hazard(s)-Fire Events and severe accident management". In this area, one or more hazard/s combined with fire should be simulated to determine the effect on the fire barriers and fire protection systems from these hazard/s. The hazard/s combination simulation should be extended to assess the impact on firefighters and emergency personnel performance when dealing with more than one hazard at the same time. Another area of research can be preforming an uncertainty and sensitivity analysis for a CANDU Fire PRA. Finally, yet importantly, some improvements can be made in a studying some specific components as fire sources and fire zones / plant areas in the OECD Fire Database.

Chapter: 10 References

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APPENDIX A: NUREG guidelines related to fire safety list

Document Identifier	Title
NUREG/CR-0152	Development and Verification of Fire Tests for Cable Systems and System Components
NUREG/CR-0381	A Preliminary Report on Fire Protection Research Program Fire Barriers and Fire Retardant Coatings Tests
NUREG/CR-0468	Nuclear Power Plant Fire Protection - Fire Barriers
NUREG/CR-0488	Nuclear Power Plant Fire Protection - Fire Detection
NUREG/CR-0596	A Preliminary Report on Fire Protection Research Program, Fire Barriers and Suppression
NUREG/CR-0636	Nuclear Power Plant Fire Protection - Ventilation
NUREG/CR-0654	Nuclear Power Plant Fire Protection - Fire-Hazards Analysis
NUREG/CR-0833	Fire Protection Research Program Corner Effects Tests
NUREG/CR-1156	Environmental Assessment of Ionization Chamber Smoke Detectors Containing Am-241
NUREG/CR-1184	Evaluation of Simulator Adequacy for the Radiation Qualification of Safety-Related Equipment
NUREG/CR-1405	The NACOM Code for Analysis of Postulated Sodium Spray Fires in LMFBRs
NUREG/CR-1552	Development and Verification of Fire Tests for Cable Systems and System Components
NUREG/CR-1614	Approaches to Acceptable Risk: A Critical Guide
NUREG/CR-1682	Electrical Insulators in a Reactor Accident Environment
NUREG/CR-1798	Acceptance and Verification For Early Warning Fire Detection Systems
NUREG/CR-1819	Development and Testing Of A Model for Fire Potential in Nuclear Power Plants
NUREG/CR-1916	A Risk Comparison
NUREG/CR-1930	Index of Risk Exposure and Risk Acceptance Criteria
NUREG/CR-2040	A Study of the Implications of Applying Quantitative Risk Criteria in the Licensing of Nuclear Power Plants in the United States
NUREG/CR-2258	Fire Risk Analysis for Nuclear Power Plants
NUREG/CR-2269	Probabilistic Models for the Behavior of Compartment Fires

NUREG/CR-2300	PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants
NUREG/CR-2321	Investigation of Fire Stop Test Parameters Final Report
NUREG/CR-2377	Test and Criteria for Fire Protection Of Cable Penetrations
NUREG/CR-2409	Requirements for Establishing Detector Siting Criteria in Fires Involving Electrical Materials
NUREG/CR-2431	Burn Mode Analysis of Horizontal Cable Tray Fires
NUREG/CR-2475	Hydrogen Combustion Characteristics Related to Reactor Accidents
NUREG/CR-2486	Final Results of the Hydrogen Igniter Experimental Program
NUREG/CR-2490	Hazards to Nuclear Power Plants from Large Liquefied Natural Gas (LNG) Spills on Water
NUREG/CR-2607	Fire Protection Research Program for the U. S. Nuclear Regulatory Commission 1975-1981
NUREG/CR-2650	Allowable Shipment Frequencies for the Transport of Toxic Gases Near Nuclear Power Plants
NUREG/CR-2658	Characteristics of Combustion Products: A Review of the Literature
NUREG/CR-2726	Light Water Reactor Hydrogen Manual
NUREG/CR-2730	Hydrogen Burn Survival: Preliminary Thermal Model and Test Results
NUREG/CR-2815	Probabilistic Safety Analysis Procedures Guide
NUREG/CR-2868	Aging Effects on Fire-Retardant Additives in Organic Materials for Nuclear Plant Applications
NUREG/CR-2927	Nuclear Power Plant Electrical Cable Damageability Experiments
NUREG/CR-3037	A Computer Code to Estimate Accidental Fire and Airborne Releases in Nuclear Fuel Cycle Facilities Radioactive
NUREG/CR-3122	Potentially Damaging Failure Modes of High- and Medium-Voltage Electrical Equipment
NUREG/CR-3139	Scenarios and Analytical Methods for UF6 Releases at NRC-Licensed Fuel Cycle Facilities
NUREG/CR-3192	Investigation of Twenty-Foot Separation Distance as a Fire Protection Method as Specified in 10 CFR 50, Appendix R
NUREG/CR-3239	COMPBRN - A Computer Code for Modeling Compartment Fires
NUREG/CR-3242	The Los Alamos National Laboratory/New Mexico State University Filter Plugging Test Facility Description and Preliminary Test Results

NUREG/CR-3263	Status Report: Correlation of Electrical Cable Failure with Mechanical Degradation
NUREG/CR-3330	Vulnerability of Nuclear Power Plant Structures to Large External Fires
NUREG/CR-3385	Measures of Risk Importance And Their Applications
NUREG/CR-3468	Hydrogen: Air: Steam Flammability Limits and Combustion Characteristics in the FITS Vessel
NUREG/CR-3493	A Review of the Limerick Generating Station Severe Accident Risk Assessment
NUREG/CR-3521	Hydrogen-Burn Survival Experiments at Fully Instrumented Test Site (FITS)
NUREG/CR-3527	Material Transport Analysis for Accident-Induced Flow in Nuclear Facilities
NUREG/CR-3532	Response of Rubber Insulation Materials to Monoenergetic Electron Irradiations
NUREG/CR-3629	The Effect of Thermal and Irradiation Aging Simulation Procedures on Polymer Properties
NUREG/CR-3638	Hydrogen-Steam Jet-Flame Facility and Experiments
NUREG/CR-3656	Evaluation of Suppression Methods for Electrical Cable Fires
NUREG/CR-3719	Detonation Calculations Using a Modified Version of CSQII: Examples for Hydrogen-Air Mixtures
NUREG/CR-3735	Accident-Induced Flow and Material Transport in Nuclear Facilities-A Literature Review
NUREG/CR-3922	Survey and Evaluation of System Interaction Events and Sources
NUREG/CR-4112	Investigation of Cable and Cable System Fire Test Parameters
NUREG/CR-4138	Data Analyses for Nevada Test Site (NTS) Premixed Combustion Tests
NUREG/CR-4229	Evaluation of Current Methodology Employed in Probabilistic Risk Assessment (PRA) of Fire Events at Nuclear Power Plants
NUREG/CR-4230	Probability-Based Evaluation of Selected Fire Protection Features in Nuclear Power Plants
NUREG/CR-4231	Evaluation of Available Data, for, Probabilistic Risk Assessments (PRA) of Fire Events at Nuclear Power Plants
NUREG/CR-4264	Investigation of High-efficiency Particulate Air Filter Plugging by Combustion Aerosols
NUREG/CR-4310	Investigation of Potential Fire-Related Damage to Safety-Related Equipment in Nuclear Power Plants

NUREG/CR-4321	Full-Scale Measurements of Smoke Transport and Deposition in Ventilation System Ductwork
NUREG/CR-4330	Review of Light Water Reactor Regulatory Requirements Identification of Regulatory Requirements That May Have Marginal Importance To Risk
NUREG/CR-4479	The Use of a Field Model to Assess Fire Behavior in Complex Nuclear Power Plant Enclosures: Present Capabilities and Future Prospects
NUREG/CR-4517	Design Features for Enhancing International Safeguards of Away-from- Reactor Dry Storage for Spent LWR Fuel
NUREG/CR-4527	An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets
NUREG/CR-4534	Analysis of Diffusion Flame Tests
NUREG/CR-4561	FIRAC User's Manual: A Computer Code to Simulate Fire Accidents in Nuclear Facilities
NUREG/CR-4566	COMPBRN III - A Computer Code for Modeling Compartment Fires
NUREG/CR-4570	Description and Testing of an Apparatus for Electrically Initiating Fires Through Simulation of a Faulty Connection
NUREG/CR-4586	User Guide for a Personal-Computer-Based Nuclear Power Plan Fire Data Base
NUREG/CR-4596	Screening Tests of Representative Nuclear Power Plant Components Exposed to Secondary Environments Created by Fires
NUREG/CR-4638	Transient Fire Environment Cable Damageability Test Results
NUREG/CR-4679	Quantitative Data on the Fire Behavior of Combustible Materials Found in Nuclear Power Plants: A Literature Review
NUREG/CR-4680	Heat and Mass Release for Some Transient Fuel Source Fires: A Test Report
NUREG/CR-4681	Enclosure Environment Characterization Testing for the Base Line Validation of Computer Fire Simulation Codes
NUREG/CR-4736	Combustion Aerosols Formed During Burning of Radioactively Contaminated Materials, Experimental Results
NUREG/CR-4829	Shipping Container Response to Severe Highway and Railway Accident Conditions
NUREG/CR-4830	MELCOR Validation and Verification: 1986 Papers
NUREG/CR-4839	Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development

NUREG/CR-4840	Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150
NUREG/CR-4855	Development and Application of a Computer Model for Large-Scale Flame Acceleration Experiments
NUREG/CR-4905	Detonability of H2-Air-Diluent Mixtures
NUREG/CR-5037	Fire Environment Determination in the LaSalle Nuclear Power Plant Control Rroom
NUREG/CR-5079	Experimental Results Pertaining to the Performance of Thermal Igniters
NUREG/CR-5233	A Computer Code for Fire Protection and Risk Analysis of Nuclear Plants
NUREG/CR-5275	FLAME Facility: The Effect of Obstacles and Transverse Venting on Flame Acceleration and Transition to Detonation for Hydrogen-Air Mixtures at Large Scale
NUREG/CR-5281	Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools
NUREG/CR-5384	A Summary of Nuclear Power Plant Fire Safety Research at Sandia National Laboratories, 1975-1987
NUREG/CR-5457	A Review of the Three Mile Island-1 Probabilistic Risk Assessment
NUREG/CR-5525	Hydrogen-Air-Diluent Detonation Study for Nuclear Reactor Safety Analyses
NUREG/CR-5546	An Investigation of the Effects of Thermal Aging on, the Fire Damageability of Electric Cables
NUREG/CR-5580	Evaluation of Generic Issue 57
NUREG/CR-5619	The Impact of Thermal Aging on the Flammability of Electric Cables
NUREG/CR-5655	Submergence and High-Temperature Steam Testing of Class 1E Electrical Cables
NUREG/CR-5669	Evaluation of Exposure Limits to Toxic Gases for Nuclear Reactor Control Room Operators
NUREG/CR-5789	Risk Evaluation for a Westinghouse PWR, Effects of Fire Protection System Actuation on Safety-Related Equipment: Evaluation of Generic Issue 57
NUREG/CR-5790	Risk Evaluation for a B&W Pressurized Water Reactor, Effects of Fire Protection System Actuation on Safety-Related Equipment: Evaluation of Generic Issue 57

NUREG/CR-5791	Risk Evaluation for a General Electric BWR, Effects of Fire Protection System Actuation on Safety-Related Equipment: Evaluation of Generic Issue 57
NUREG/CR-6017	Fire Modeling of the Heiss Dampf Reaktor Containment
NUREG/CR-6042	Perspectives on Reactor Safety
NUREG/CR-6095	Aging, Loss-of-Coolant Accident (LOCA), and High Potential Testing of Damaged Cables
NUREG/CR-6173	A Summary of the Fire Testing Program at the German HDR Test Facility
NUREG/CR-6213	High-Temperature Hydrogen-Air- Steam Detonation Experiments in the BNL Small-Scale Development Apparatus
NUREG/CR-6220	An Assessment of Fire Vulnerability for Aged Electrical Relays
NUREG/CR-6358	Assessment of United States Industry Structural Codes and Standards for Application to Advanced Nuclear Power Reactors
NUREG/CR-6384	Literature Review of Environmental Qualification of Safety-Related Electric Cables
NUREG/CR-6406	Environmental Testing of an Experimental Digital Safety Channel
NUREG/CR-6410	Nuclear Fuel Cycle Facility Accident Analysis Handbook
NUREG/CR-6476	Circuit Bridging of Components by Smoke
NUREG/CR-6479	Technical Basis for Environmental Qualification of Microprocessor-Based Safety-Related Equipment in Nuclear Power Plants
NUREG/CR-6509	The Effect of Initial Temperature on Flame Acceleration and Deflagration-to-Detonation Transition Phenomenon
NUREG/CR-6524	The Effect of Lateral Venting on Deflagration-to-Detonation Transition in Hydrogen-Air-Steam Mixtures at Various Initial Temperatures
NUREG/CR-6530	Deliberate Ignition of Hydrogen-Air-Steam Mixtures in Condensing Steam Environments
NUREG/CR-6543	Effects of Smoke on Functional Circuits
NUREG/CR-6544	A Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences
NUREG/CR-6597	Results and Insights on the Impact of Smoke on Digital Instrumentation and Control
NUREG/CR-6681	Ampacity Derating and Cable Functionality for Raceway Fire Barriers
NUREG/CR-6738	Risk Methods Insights Gained from Fire Incidents

NUREG/CR-6752	A Comparative Analysis of Special Treatment Requirements for Systems, Structures, and Components (SSCs) of Nuclear Power Plants with Commercial Requirements of Non-Nuclear Power Plants

APPENDIX B: Fuel Survey Sample Calculation

The *Instrumentation Room X* was selected as the sample fuel survey following NFPA 557 combination method. There are a total of 74 Instrumentation Rooms listed as FSSA rooms in the fuel survey.

Fire Zone Description: The room contains instrumentation associated with the main circulation system.

Floor area: 273.1 m^2
Height: 4.6 m
Ceiling: Poured concrete
Floor: Poured concrete
Wall: Poured concrete
Penetrations: Sealed cable tray penetrations are located in the south wall. There is an HVAC penetration in the ceiling at the northwest corner of the room. Sealed cable penetrations are located in the east and west walls

Suppression Systems: Automatic suppression is not provided in the fire zone
Detection Systems: Automatic detection is not provided in the fire zone
Emergency Lighting: Emergency lighting is provided in the area
Portable Extinguishers & Fire Hose Cabinets: Portable extinguishers and a fire hose station are located in adjacent fire zone. These are adequate means of manual suppression for the transient fires postulated for this fire zone

Ventilation: Ventilation is provided
Drainage: Drainage is not provided in the fire zone
Potential Hazards: Transient Combustibles
High Energy Arcing Faults (HEAF): HEAF hazards are not present in the area.
Potential Ignition Sources: Ventilated Electrical Panel (Electrical)

Distance between nearest combustibles to FSSA cable: 5 meters

Types and Quantity of Combustibles:

Combustible Type	Quantity	Units of Measure
PVC Cable Tray- 2" Diameter	27.0	Meters
Plastics	15.65	Kg

PVC Cable - 2" Diameter:

Calorific Value: 30.8 MJ/Kg
Density: 1380 Kg/m^3
Volume: $V = \pi r^2 h = \pi\ x\ (0.025\ ^2)\ x\ (27\ m) = 0.053\ m^3$
Weight = 1380 Kg/m^3 x 0.053 m^3 = 111.09 Kg
Total MJ = 30.8 MJ/Kg x 111.09 Kg = 3,421.71 MJ (using Inventory method)

Plastics:

Heat of combustion: 46.5 KJ/Kg
Weight = 15.65 Kg
Total MJ = 46.5 MJ/Kg x 15.65 Kg = 727.69 MJ (using Weighing method)

Using Combination method

Total Combustibles: 3,421.71 MJ + 727.69 MJ = 4149.4 MJ

Fuel Density: 4149.4 MJ / 273.1 m^2 = 15.19 MJ/m^2

There are 74 Instrumentation Rooms:

Maximum Fuel Density: 248.34 MJ/m^2 (this is the maximum fuel density for a fire zone in all 74 fire zones)
Average Fuel Density: 57.77 MJ/m^2 (this the sum of all 74 fire zones fuel densities divided by 74)
Minimum Fuel Density: 13.54 MJ/m^2 (this is the minimum fuel density for a fire zone in all 74 fire zones)

APPENDIX C: FDS Simulation Input File

All walls, floors, and ceilings of the nine simulations are made of concrete. Concrete properties:

Thermal Conductivity = 0.001 KW/mK,
Density= 2000 kg/m^3,
Specific heat 0.88 kJ/kg,
Wall thickness 0.3 m;

The description of the thermal properties of the cables in the FDS input file is given below:

```
&MATL ID='Sheath',
SPECIFIC_HEAT=1.0,
CONDUCTIVITY=0.05,
DENSITY=1501.0/
&MATL ID='Filler',
SPECIFIC_HEAT=3.0,
CONDUCTIVITY=0.15,
DENSITY=950.0/
&MATL ID='Insulation',
SPECIFIC_HEAT=3.5,
CONDUCTIVITY=0.2,
DENSITY=1039.0/

&SURF ID='Cable',
COLOR='GRAY 20',
HRRPUA=589.0,
RAMP_Q='Cable_RAMP_Q',
IGNITION_TEMPERATURE=205.0,
BURN_AWAY=.TRUE.,
MATL_ID(1,1)='Sheath',
MATL_ID(2,1:2)='Filler','Insulation',
MATL_ID(3,1)='Sheath',
MATL_MASS_FRACTION(1,1)=1.0,
MATL_MASS_FRACTION(2,1:2)=0.5,0.5,
MATL_MASS_FRACTION(3,1)=1.0,
THICKNESS(1:3)=0.025,0.01,0.025/
```

500 MJ of power cable combustible load FDS input:

```
&RAMP ID='Cable_RAMP_Q', T=0.0, F=0.0/
&RAMP ID='Cable_RAMP_Q', T= 51.0, F=1.0/
&RAMP ID='Cable_RAMP_Q', T=1052.0, F=1.0/
&RAMP ID='Cable_RAMP_Q', T=1103.0, F=0.0/
```

5,000 MJ of power cable combustible load FDS input:

```
&RAMP ID='Cable_RAMP_Q', T=0.0, F=0.0/
&RAMP ID='Cable_RAMP_Q', T=111.0, F=1.0/
&RAMP ID='Cable_RAMP_Q', T=2002.0, F=1.0/
&RAMP ID='Cable_RAMP_Q', T=2113.0, F=0.0/
```

Simulation 1: ***(500 MJ) 38 m²***

- Simulation conditions:
  - Domain, room dimensions XYZ 5.8 meters x 6.5 meters x 4.2 meters;
  - Thermocouple & heat flux sensor XYZ 3.8, 3.7, 3.0;
  - Location of cable (First Tray) XYZ 3.0, 5.5, 0.9 to XYZ 3.33, 5.86, 1.02 (Second Tray) XYZ 3.0, 5.5, 1.2 to XYZ 3.33, 5.86, 1.32;
  - Distance between Thermocouple/heat flux sensor and fire is
  - $\{(3.0 – 3.8)2 + (5.5 - 3.7)2\}^{1/2} = 1.97$ m;
  - Door open, vent XYZ 2.0, 3.0, 0.1 to XYZ 3.7, 3.0, 2.6
  - Mesh size # of elements 40 x 65 x 42 = 109,200; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 3.0, 5.5, 0.1.

Simulation 1.1: ***(500 MJ) 26 m²***

- Simulation conditions:
  - Domain, room dimensions XYZ 4.0 meters x 6.5 meters x 4.2 meters;
  - Thermocouple & heat flux sensor XYZ 3.8, 3.7, 3.0;
  - Location of cable (First Tray) XYZ 3.0, 5.5, 0.9 to XYZ 3.33, 5.86, 1.02 (Second Tray) XYZ 3.0, 5.5, 1.2 to XYZ 3.33, 5.86, 1.32;
  - Distance between Thermocouple/heat flux sensor and fire is
  - $\{(3.0 – 3.8)2 + (5.5 - 3.7)2\}^{1/2} = 1.97$ m;
  - Door open, vent XYZ 2.0, 3.0, 0.1 to XYZ 3.7, 3.0, 2.6
  - Mesh size # of elements 40 x 65 x 42 = 109,200; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 3.0, 5.5, 0.1.

Simulation 1.2: ***(500 MJ) 50 m²***

- Simulation conditions:
  - Domain, room dimensions XYZ 7.7 meters x 6.5 meters x 4.2 meters;
  - Thermocouple & heat flux sensor XYZ 3.8, 3.7, 3.0;
  - Location of cable (First Tray) XYZ 3.0, 5.5, 0.9 to XYZ 3.33, 5.86, 1.02 (Second Tray) XYZ 3.0, 5.5, 1.2 to XYZ 3.33, 5.86, 1.32;
  - Distance between Thermocouple/heat flux sensor and fire is

- $\{(3.0 – 3.8)2 + (5.5 - 3.7)2\}^{1/2} = 1.97$ m;
- Door open, vent XYZ 2.0, 3.0, 0.1 to XYZ 3.7, 3.0, 2.6
- Mesh size # of elements 40 x 65 x 42 = 109,200; and
- Grid size: XYZ 0.1 m x 0.1 m x 0.1 m

- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 3.0, 5.5, 0.1.

Simulation 2: ***(500 MJ) 38 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 5.8 meters x 6.5 meters x 4.2 meters;
  - Thermocouple & heat flux sensor XYZ 0.92, 1.0, 3.0;
  - Location of combustibles (First Tray) XYZ 3.0, 5.5, 0.9 to XYZ 3.33, 5.86, 1.02
    (Second Tray) XYZ 3.0, 5.5, 1.2 to XYZ 3.33, 5.86, 1.32;
  - Distance between Thermocouple/heat flux sensor and fire is $\{(3.0 – 0.9)^2 + (5.5 – 1.0)^2\}^{1/2} = 4.96$ m;
  - Door open, vent XYZ 2.0, 3.0, 0.1 to XYZ 3.7, 3.0, 2.6
  - Mesh size # of elements 40 x 65 x 42 = 109,200; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 3.0, 5.5, 0.1.

Simulation 2.1: ***(500 MJ) 26 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 4.0 meters x 6.5 meters x 4.2 meters;
  - Thermocouple & heat flux sensor XYZ 0.92, 1.0, 3.0;
  - Location of combustibles (First Tray) XYZ 3.0, 5.5, 0.9 to XYZ 3.33, 5.86, 1.02
    (Second Tray) XYZ 3.0, 5.5, 1.2 to XYZ 3.33, 5.86, 1.32;
  - Distance between Thermocouple/heat flux sensor and fire is $\{(3.0 – 0.9)^2 + (5.5 – 1.0)^2\}^{1/2} = 4.96$ m;
  - Door open, vent XYZ 2.0, 3.0, 0.1 to XYZ 3.7, 3.0, 2.6
  - Mesh size # of elements 40 x 65 x 42 = 109,200; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 3.0, 5.5, 0.1.

Simulation 2.2: ***(500 MJ) 50 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 7.7 meters x 6.5 meters x 4.2 meters;

- Thermocouple & heat flux sensor XYZ 0.92, 1.0, 3.0;
- Location of combustibles (First Tray) XYZ 3.0, 5.5, 0.9 to XYZ 3.33, 5.86, 1.02
  (Second Tray) XYZ 3.0, 5.5, 1.2 to XYZ 3.33, 5.86, 1.32;
- Distance between Thermocouple/heat flux sensor and fire is $\{(3.0 - 0.9)^2 + (5.5 - 1.0)^2\}^{1/2} = 4.96$ m;
- Door open, vent XYZ 2.0, 3.0, 0.1 to XYZ 3.7, 3.0, 2.6
- Mesh size # of elements 40 x 65 x 42 = 109,200; and
- Grid size: XYZ 0.1 m x 0.1 m x 0.1 m

- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 3.0, 5.5, 0.1.

Simulation 3: ***(500 MJ) 74 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 7.4 meters x 10.0 meters x 5.3 meters;
  - Thermocouple & heat flux sensor XYZ 3.8, 6.3, 4.0;
  - Location of combustibles (First Tray) XYZ 4.6, 4.5, 0.9 to XYZ 4.93, 4.86, 1.02
    (Second Tray) XYZ 4.6, 4.5, 1.2 to XYZ 4.93, 4.86, 1.32;
  - Distance between Thermocouple/heat flux sensor and fire is $\{(3.8 - 4.6)^2 + (6.3 - 4.5)^2\}^{1/2} = 1.97$ m;
  - Door open, vent XYZ 3.0, 5.0, 0.1 to XYZ 4.7, 5.0, 2.6
  - Mesh size # of elements 74 x 100 x 53 = 392,200; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 4.6, 4.5, 0.1.

Simulation 3.1: ***(500 MJ) 50 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 7.4 meters x 6.7 meters x 5.3 meters;
  - Thermocouple & heat flux sensor XYZ 3.8, 6.3, 4.0;
  - Location of combustibles (First Tray) XYZ 4.6, 4.5, 0.9 to XYZ 4.93, 4.86, 1.02
    (Second Tray) XYZ 4.6, 4.5, 1.2 to XYZ 4.93, 4.86, 1.32;
  - Distance between Thermocouple/heat flux sensor and fire is $\{(3.8 - 4.6)^2 + (6.3 - 4.5)^2\}^{1/2} = 1.97$ m;
  - Door open, vent XYZ 3.0, 5.0, 0.1 to XYZ 4.7, 5.0, 2.6
  - Mesh size # of elements 74 x 100 x 53 = 392,200; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 4.6, 4.5, 0.1.

Simulation 3.2: ***(500 MJ) 100 m²***

- Simulation conditions:
  - Domain, room dimensions XYZ 10.0 meters x 10.0 meters x 5.3 meters;
  - Thermocouple & heat flux sensor XYZ 6.2, 9.7, 5.0;
  - Location of combustibles (First Tray) XYZ 7.1, 7.9, 0.9 to XYZ 7.43, 8.26, 1.02
    (Second Tray) XYZ 7.1, 7.9, 1.2 to XYZ 7.43, 8.26, 1.32
  - Distance between Thermocouple/heat flux sensor and fire is $\{(6.2 - 7.1)^2 + (9.7 - 7.9)^2\}^{1/2} = 2.0$ m
  - Door open, vent XYZ 9.0, 5.0, 0.1 to XYZ 10.7, 5.0, 2.6
  - Mesh size # of elements 187 x 100 x 61 = 1,140,700; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 7.1, 7.9, 0.1.

Simulation 6: ***(500 MJ) 187 m²***

- Simulation conditions:
  - Domain, room dimensions XYZ 18.7 meters x 10.0 meters x 6.1 meters;
  - Thermocouple & heat flux sensor XYZ 6.2, 9.7, 5.0;
  - Location of combustibles (First Tray) XYZ 7.1, 7.9, 0.9 to XYZ 7.43, 8.26, 1.02
    (Second Tray) XYZ 7.1, 7.9, 1.2 to XYZ 7.43, 8.26, 1.32
  - Distance between Thermocouple/heat flux sensor and fire is $\{(6.2 - 7.1)^2 + (9.7 - 7.9)^2\}^{1/2} = 2.0$ m
  - Door open, vent XYZ 9.0, 5.0, 0.1 to XYZ 10.7, 5.0, 2.6
  - Mesh size # of elements 187 x 100 x 61 = 1,140,700; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 7.1, 7.9, 0.1.

Simulation 6.1: ***(500 MJ) 100 m²***

- Simulation conditions:
  - Domain, room dimensions XYZ 10.0 meters x 10.0 meters x 6.1 meters;
  - Thermocouple & heat flux sensor XYZ 6.2, 9.7, 5.0;
  - Location of combustibles (First Tray) XYZ 7.1, 7.9, 0.9 to XYZ 7.43, 8.26, 1.02
    (Second Tray) XYZ 7.1, 7.9, 1.2 to XYZ 7.43, 8.26, 1.32
  - Distance between Thermocouple/heat flux sensor and fire is $\{(6.2 - 7.1)^2 + (9.7 - 7.9)^2\}^{1/2} = 2.0$ m
  - Door open, vent XYZ 9.0, 5.0, 0.1 to XYZ 10.7, 5.0, 2.6
  - Mesh size # of elements 187 x 100 x 61 = 1,140,700; and

    - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
    - Burner: HRRPUA 233.3 KW/m^2;
    - Burner Location XYZ 7.1, 7.9, 0.1.

Simulation 9: ***(500 MJ) 571 m^2***

- Simulation conditions:
    - Domain, room dimensions XYZ 25.0 meters x 23.0 meters x 7.0 meters;
    - Thermocouple & heat flux sensor XYZ 6.1, 9.3, 6.0;
    - Location of cable (First Tray) XYZ 6.9, 11.1, 0.9 to XYZ 7.23, 11.46, 1.02
      (Second Tray) XYZ 6.9, 11.1, 1.2 to XYZ 7.23, 11.46, 1.32;
    - Distance between Thermocouple/heat flux sensor and fire is $\{(3.8 – 4.6)^2 + (6.3 – 4.5)^2\}^{1/2} = 1.97$ m;
    - Door open, vent XYZ 5.0, 13.8, 0.1 to XYZ 6.7, 13.8, 2.6
    - Mesh size # of elements 250 x 230 x 70 = 4,025,000; and
    - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
    - Burner: HRRPUA 233.3 KW/m^2;
    - Burner Location XYZ 6.9, 11.1, 0.1;

Simulation 9.1: ***(500 MJ) 400 m^2***

- Simulation conditions:
    - Domain, room dimensions XYZ 20.0 meters x 20.0 meters x 7.0 meters;
    - Thermocouple & heat flux sensor XYZ 6.1, 9.3, 6.0;
    - Location of cable (First Tray) XYZ 6.9, 11.1, 0.9 to XYZ 7.23, 11.46, 1.02
      (Second Tray) XYZ 6.9, 11.1, 1.2 to XYZ 7.23, 11.46, 1.32;
    - Distance between Thermocouple/heat flux sensor and fire is $\{(3.8 – 4.6)^2 + (6.3 – 4.5)^2\}^{1/2} = 1.97$ m;
    - Door open, vent XYZ 5.0, 13.8, 0.1 to XYZ 6.7, 13.8, 2.6
    - Mesh size # of elements 250 x 230 x 70 = 4,025,000; and
    - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
    - Burner: HRRPUA 233.3 KW/m^2;
    - Burner Location XYZ 6.9, 11.1, 0.1;

Simulation 13: ***(5,000 MJ) 38 m^2***

- Simulation conditions:
    - Domain, room dimensions XYZ 5.8 meters x 6.5 meters x 4.2 meters;
    - Thermocouple & heat flux sensor XYZ 0.6, 0.8, 3.0;
    - Location of cable (First Tray) XYZ 4.8, 3.5, 0.9 to XYZ 6.44, 3.86, 1.14
      (Second Tray) XYZ 4.8, 3.5, 1.4 to XYZ 6.44, 3.86, 1.64;

- Distance between Thermocouple/heat flux sensor and fire is $\{(0.6 – 4.8)^2 + (0.8 – 3.5)^2\}^{1/2} = 5$ m;
- Door open, vent XYZ 2.0, 3.0, 0.1 to XYZ 3.7, 3.0, 2.6
- Mesh size # of elements 65 x 40 x 42 = 109,200; and
- Grid size: XYZ 0.1 m x 0.1 m x 0.1 m

- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 4.8, 3.5, 0.1.

Simulation 13.2: ***(5,000 MJ) 50 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 7.7 meters x 6.5 meters x 4.2 meters;
  - Thermocouple & heat flux sensor XYZ 0.6, 0.8, 3.0;
  - Location of cable (First Tray) XYZ 4.8, 3.5, 0.9 to XYZ 6.44, 3.86, 1.14
    (Second Tray) XYZ 4.8, 3.5, 1.4 to XYZ 6.44, 3.86, 1.64;
  - Distance between Thermocouple/heat flux sensor and fire is $\{(0.6 – 4.8)^2 + (0.8 – 3.5)^2\}^{1/2} = 5$ m;
  - Door open, vent XYZ 2.0, 3.0, 0.1 to XYZ 3.7, 3.0, 2.6
  - Mesh size # of elements 65 x 40 x 42 = 109,200; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 4.8, 3.5, 0.1.

Simulation 16: ***(5,000 MJ) 74 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 10.0 meters x 7.4 meters x 5.3 meters;
  - Thermocouple & heat flux sensor XYZ 0.4, 0.7, 4.0;
  - Location of cable (First Tray) XYZ 4.0, 6.7, 0.9 to XYZ 5.64, 7.06, 1.14
    (Second Tray) XYZ 4.0, 6.7, 1.4 to XYZ 5.64, 7.06, 1.64;
  - Distance between Thermocouple/heat flux sensor and fire is $\{(0.4 – 4.0)^2 + (0.7 – 6.7)^2\}^{1/2} = 7$ m;
  - Door open, vent XYZ 3.0, 5.0, 0.1 to XYZ 4.7, 5.0, 2.6
  - Mesh size # of elements 10 x 74 x 53 = 392,200; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 4.0, 6.7, 0.1.

Simulation 16.2: ***(5,000 MJ) 100 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 10.0 meters x 10.0 meters x 5.3 meters;
  - Thermocouple & heat flux sensor XYZ 0.4, 0.7, 4.0;
  - Location of cable (First Tray) XYZ 4.0, 6.7, 0.9 to XYZ 5.64, 7.06, 1.14

(Second Tray) XYZ 4.0, 6.7, 1.4 to XYZ 5.64, 7.06, 1.64;
    - Distance between Thermocouple/heat flux sensor and fire is $\{(0.4 – 4.0)^2 + (0.7 – 6.7)^2\}^{1/2} = 7$ m;
    - Door open, vent XYZ 3.0, 5.0, 0.1 to XYZ 4.7, 5.0, 2.6
    - Mesh size # of elements 10 x 74 x 53 = 392,200; and
    - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
    - Burner: HRRPUA 233.3 KW/m^2;
    - Burner Location XYZ 4.0, 6.7, 0.1.

Simulation 19: ***(5,000 MJ) 187 m^2***

- Simulation conditions:
    - Domain, room dimensions XYZ 18.7 meters x 10.0 meters x 6.1 meters;
    - Thermocouple & heat flux sensor XYZ 12.2, 9.7, 5.0;
    - Location of cable (First Tray) XYZ 4.26, 5.0, 0.9 to XYZ 5.9, 5.36, 1.14 (Second Tray) XYZ 4.26, 5.0, 1.4 to XYZ 5.9, 5.36, 1.64;
    - Distance between Thermocouple/heat flux sensor and fire is $\{(12.2 – 5.9)^2 + (9.7 – 6.64)^2\}^{1/2} = 7$ m;
    - Door open, vent XYZ 9.0, 5.0, 0.1 to XYZ 10.7, 5.0, 2.6
    - Mesh size # of elements 187 x 100 x 61 = 1,140,700; and
    - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
    - Burner: HRRPUA 233.3 KW/m^2;
    - Burner Location XYZ 5.0, 5.2, 0.1

Simulation 19.1: ***(5,000 MJ) 100 m^2***

- Simulation conditions:
    - Domain, room dimensions XYZ 10.0 meters x 10.0 meters x 6.1 meters;
    - Thermocouple & heat flux sensor XYZ 2.2, 9.7, 5.0;
    - Location of cable (First Tray) XYZ 4.26, 5.0, 0.9 to XYZ 5.9, 5.36, 1.14 (Second Tray) XYZ 4.26, 5.0, 1.4 to XYZ 5.9, 5.36, 1.64;
    - Distance between Thermocouple/heat flux sensor and fire is $\{(2.2 – 5.9)^2 + (9.7 – 6.64)^2\}^{1/2} = 7$ m;
    - Door open, vent XYZ 8.0, 5.0, 0.1 to XYZ 9.7, 5.0, 2.6
    - Mesh size # of elements 187 x 100 x 61 = 1,140,700; and
    - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
    - Burner: HRRPUA 233.3 KW/m^2;
    - Burner Location XYZ 5.0, 5.2, 0.1

Simulation 19.2: ***(5,000 MJ) 400 m^2***

- Simulation conditions:
    - Domain, room dimensions XYZ 20.0 meters x 20.0 meters x 6.1 meters;

- Thermocouple & heat flux sensor XYZ 12.2, 9.7, 5.0;
- Location of cable (First Tray) XYZ 4.26, 5.0, 0.9 to XYZ 5.9, 5.36, 1.14 (Second Tray) XYZ 4.26, 5.0, 1.4 to XYZ 5.9, 5.36, 1.64;
- Distance between Thermocouple/heat flux sensor and fire is $\{(12.2 - 5.9)^2 + (9.7 - 6.64)^2\}^{1/2} = 7$ m;
- Door open, vent XYZ 9.0, 5.0, 0.1 to XYZ 10.7, 5.0, 2.6
- Mesh size # of elements 187 x 100 x 61 = 1,140,700; and
- Grid size: XYZ 0.1 m x 0.1 m x 0.1 m

- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 5.0, 5.2, 0.1

Simulation 22: ***(5,000 MJ) 571 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 25.0 meters x 23.0 meters x 7.0 meters;
  - Thermocouple & heat flux sensor XYZ 12.2, 9.7, 5.0;
  - Location of cable (First Tray) XYZ 4.26, 5.0, 0.9 to XYZ 5.9, 5.36, 1.14 (Second Tray) XYZ 4.26, 5.0, 1.4 to XYZ 5.9, 5.36, 1.64;
  - Distance between Thermocouple/heat flux sensor and fire is $\{(12.2 - 5.9)^2 + (9.7 - 6.5)^2\}^{1/2} = 7$ m;
  - Door open, vent XYZ 5.0, 13.8, 0.1 to XYZ 6.7, 13.8, 2.6
  - Mesh size # of elements 250 x 230 x 70 = 4,025,000; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 5.0, 5.2, 0.1

Simulation 22.1: ***(5,000 MJ) 400 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 20.0 meters x 20.0 meters x 7.0 meters;
  - Thermocouple & heat flux sensor XYZ 12.2, 9.7, 5.0;
  - Location of cable (First Tray) XYZ 4.26, 5.0, 0.9 to XYZ 5.9, 5.36, 1.14 (Second Tray) XYZ 4.26, 5.0, 1.4 to XYZ 5.9, 5.36, 1.64;
  - Distance between Thermocouple/heat flux sensor and fire is $\{(12.2 - 5.9)^2 + (9.7 - 6.5)^2\}^{1/2} = 7$ m;
  - Door open, vent XYZ 5.0, 13.8, 0.1 to XYZ 6.7, 13.8, 2.6
  - Mesh size # of elements 250 x 230 x 70 = 4,025,000; and
  - Grid size: XYZ 0.1 m x 0.1 m x 0.1 m
- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 5.0, 5.2, 0.1

Simulation 22.2: ***(5,000 MJ) 1200 m^2***

- Simulation conditions:
  - Domain, room dimensions XYZ 25.0 meters x 48.0 meters x 7.0 meters;

- Thermocouple & heat flux sensor XYZ 12.2, 9.7, 5.0;
- Location of cable (First Tray) XYZ 4.26, 5.0, 0.9 to XYZ 5.9, 5.36, 1.14 (Second Tray) XYZ 4.26, 5.0, 1.4 to XYZ 5.9, 5.36, 1.64;
- Distance between Thermocouple/heat flux sensor and fire is $\{(12.2 – 5.9)^2 + (9.7 – 6.5)^2\}^{1/2} = 7$ m;
- Door open, vent XYZ 5.0, 13.8, 0.1 to XYZ 6.7, 13.8, 2.6
- Mesh size # of elements 250 x 230 x 70 = 4,025,000; and
- Grid size: XYZ 0.1 m x 0.1 m x 0.1 m

- Boundary conditions:
  - Burner: HRRPUA 233.3 KW/m^2;
  - Burner Location XYZ 5.0, 5.2, 0.1

APPENDIX D: HRR and Heat Flux Output

Simulations 1, 2, 3, 6 & 9 output:

The Fire Development curves below show a fuel controlled fire with 500 MJ of fuel as demonstrated in Simulations 1, 2, 3, 6, and 9. The energy level increases as more fuel becomes involved in the fire (up to 5 minutes), until all of the available fuel is involved. As the fuel is burned away, the energy level begins to decay (22 minutes to 30 minutes). Figures 37 – 49 below show the maximum temperature measured at the thermocouple, the maximum heat flux measured at the heat flux sensor, and the maximum HRR reached for Simulations 1, 2, 3, 6 & 9.

Figure 36. Simulation 1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

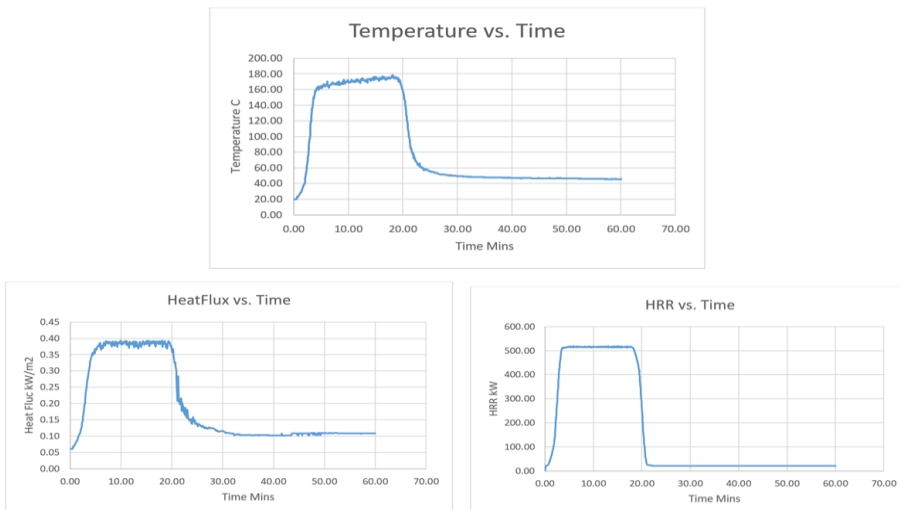


Figure 37. Simulation 1.1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

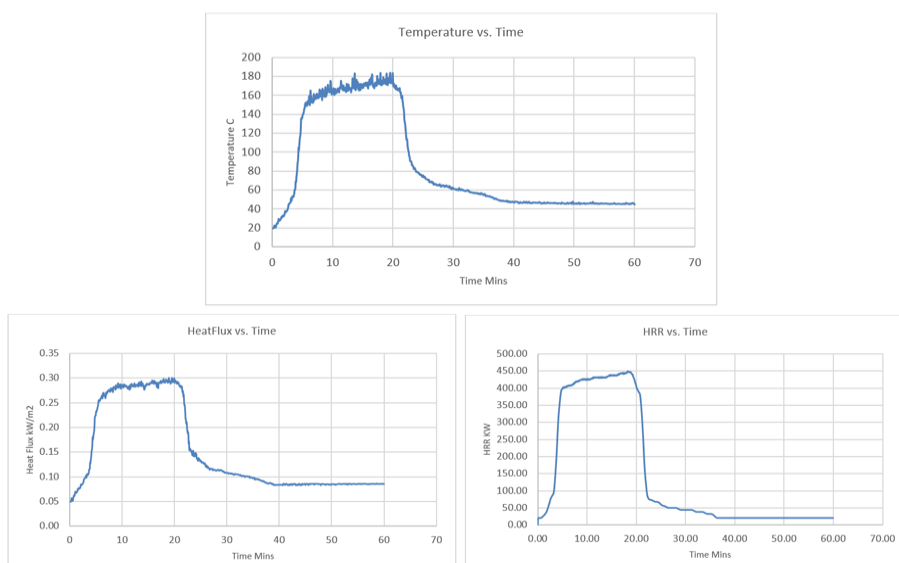


Figure 38. Simulation 1.2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

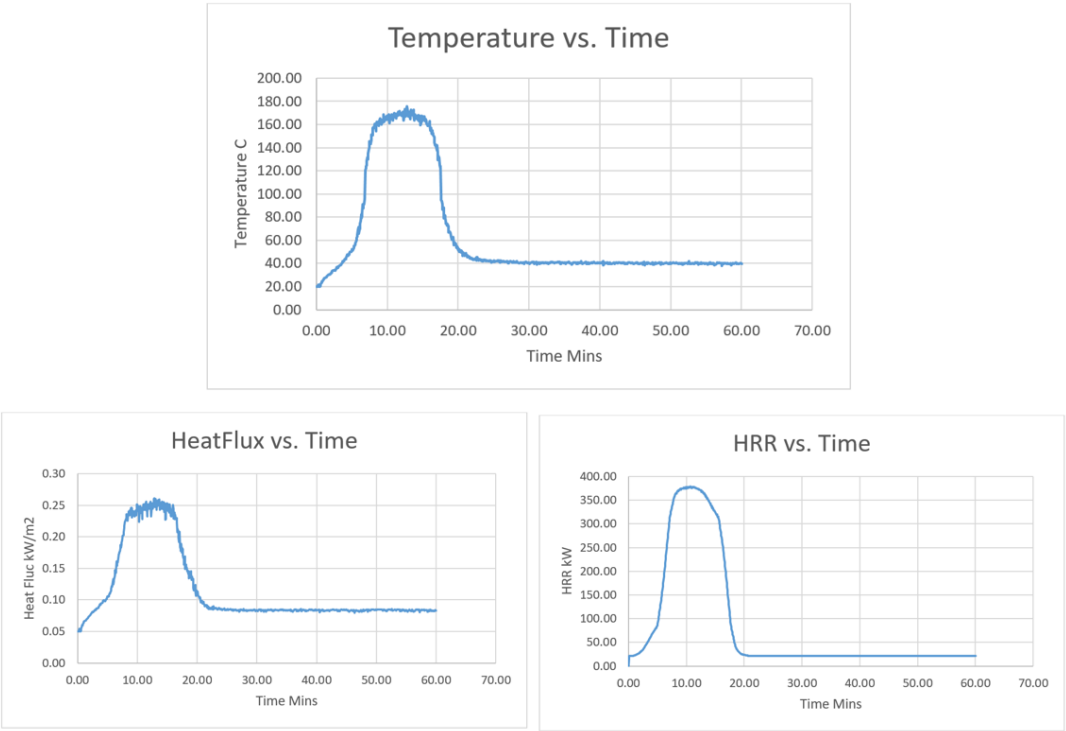


Figure 39. Simulation 2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

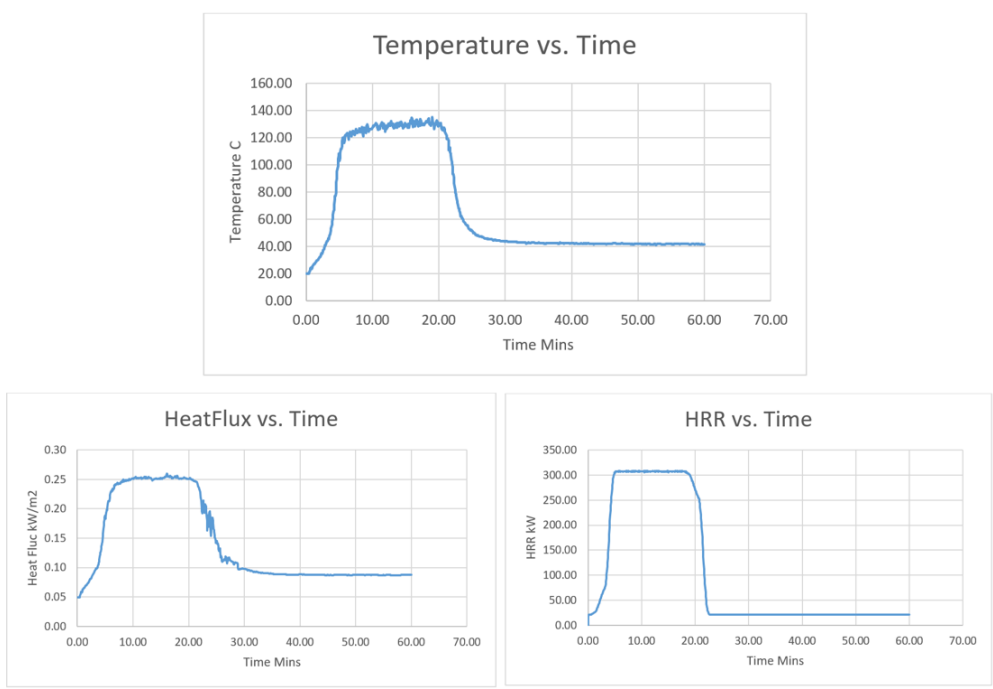


Figure 40. Simulation 2.1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

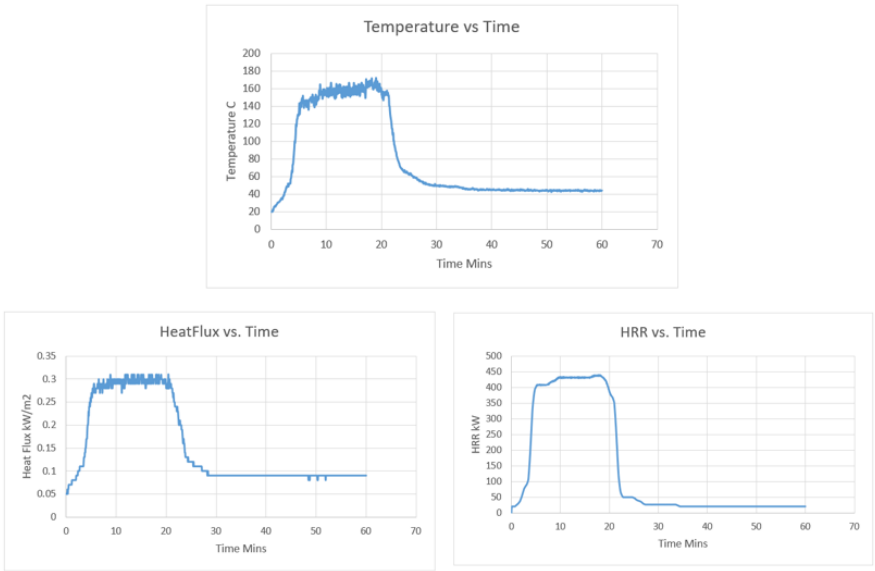


Figure 41. Simulation 2.2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

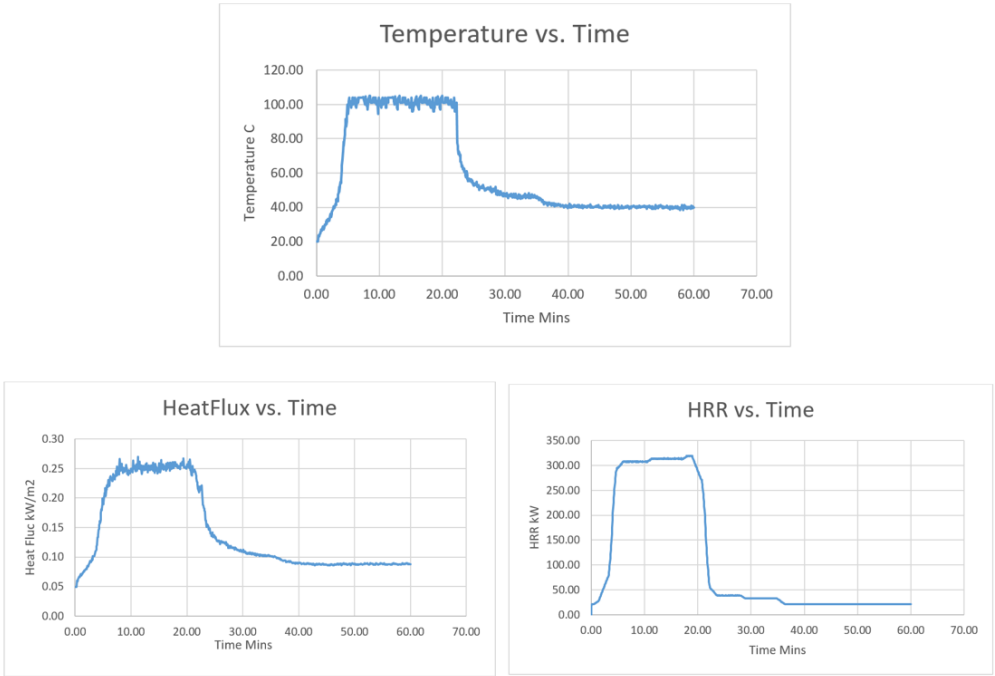


Figure 42. Simulation 3 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

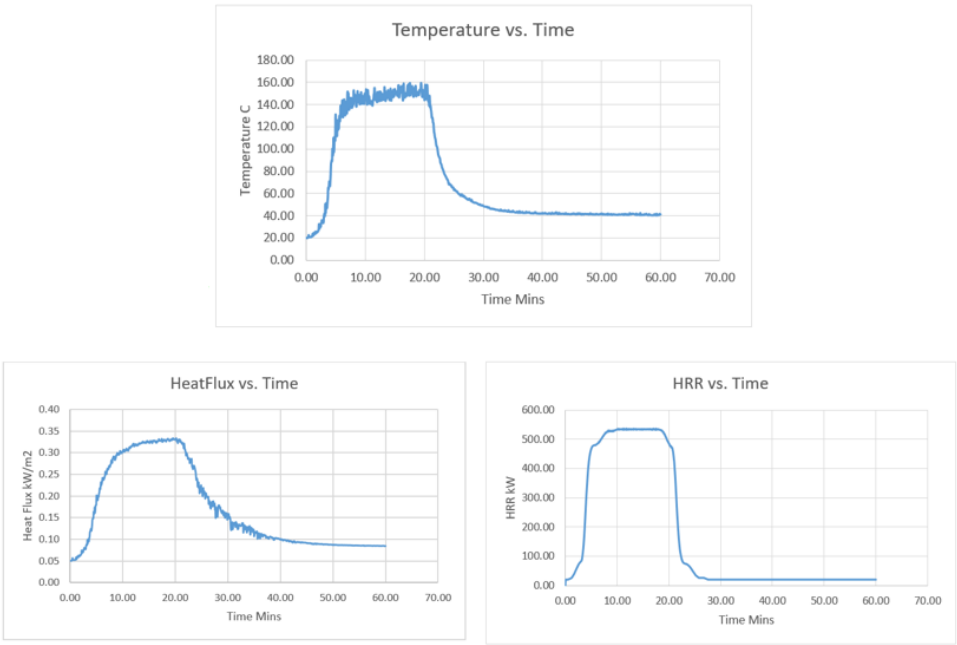


Figure 43. Simulation 3.1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

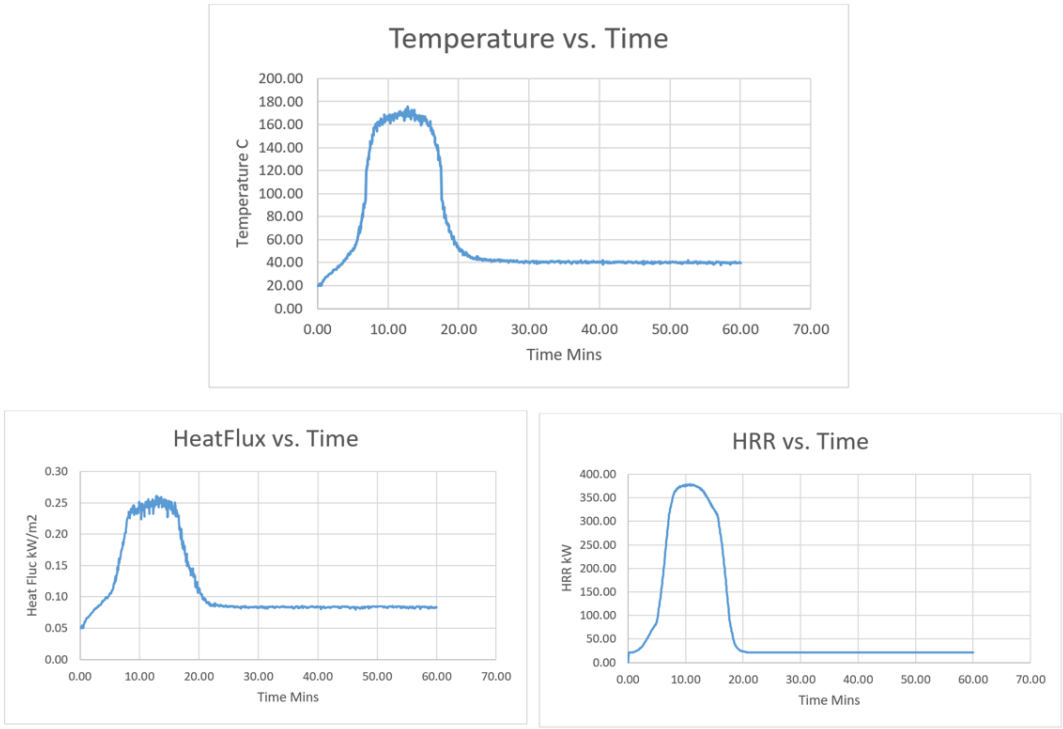


Figure 44. Simulation 3.2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

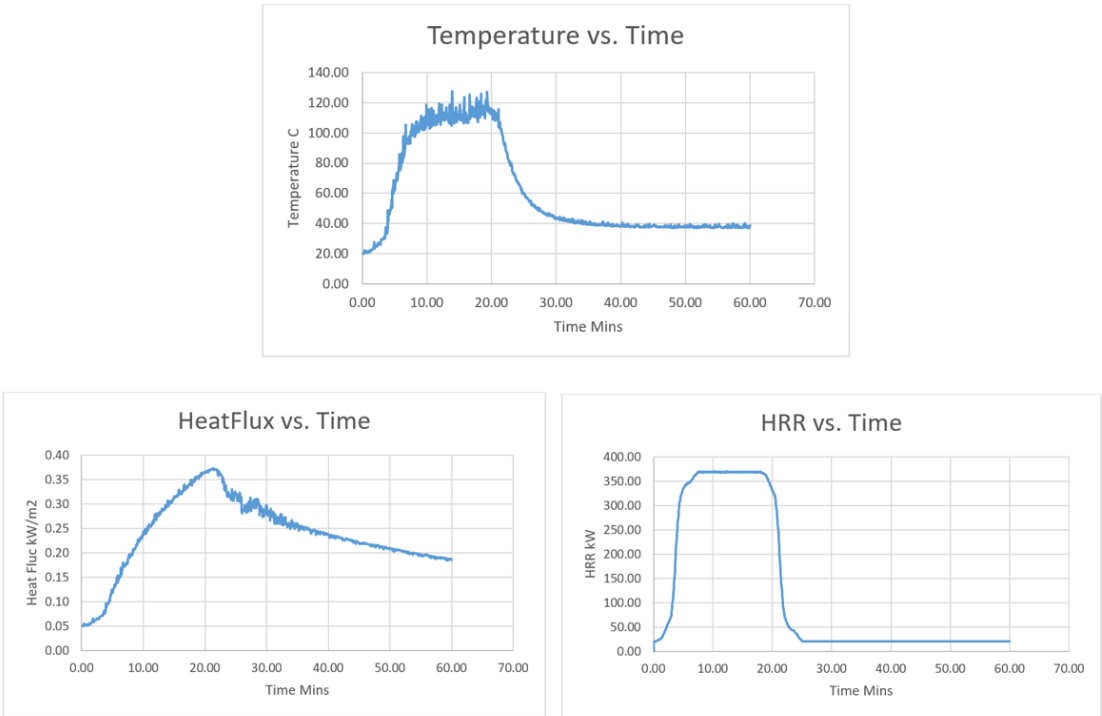


Figure 45. Simulation 6 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

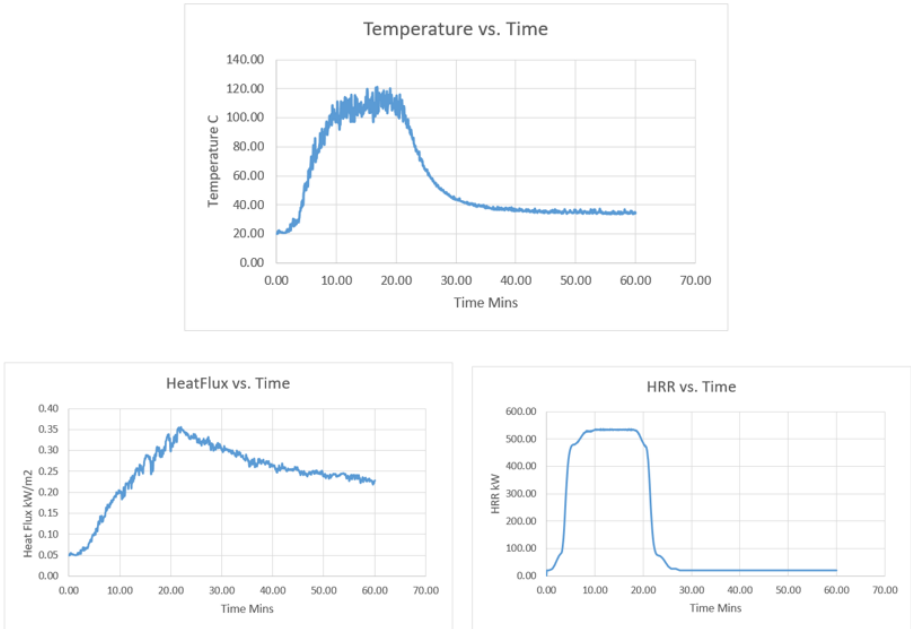


Figure 46. Simulation 6.1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

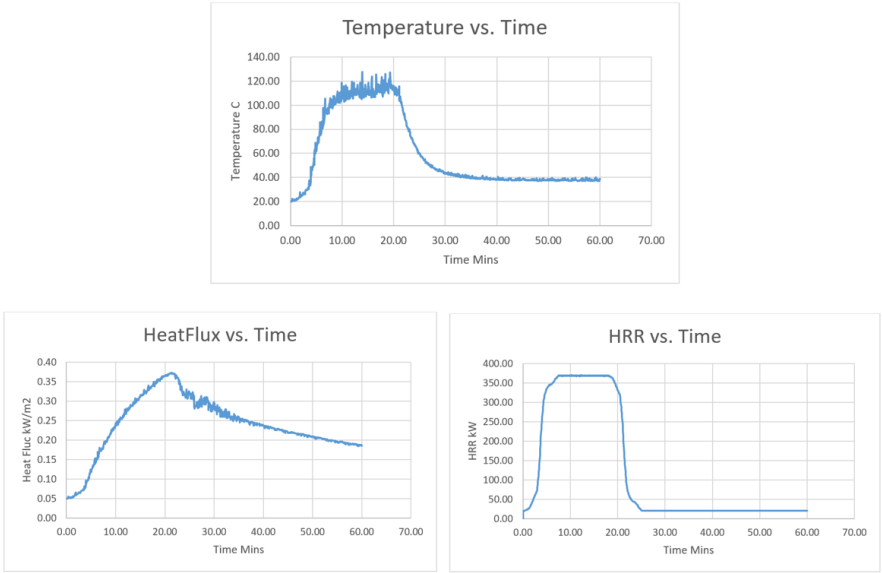


Figure 47. Simulation 9 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

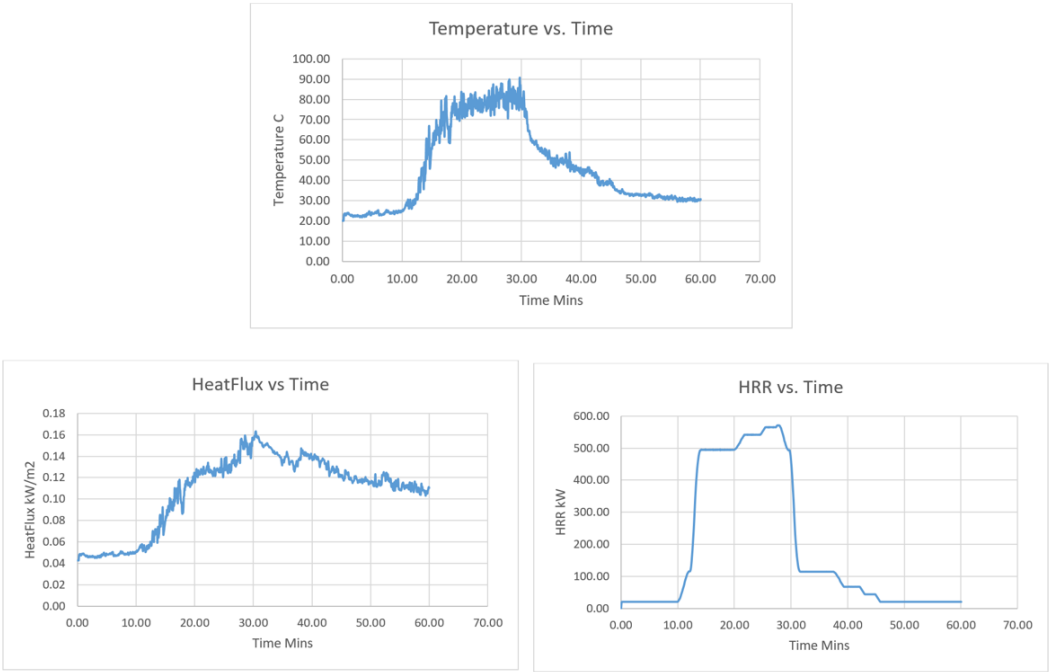
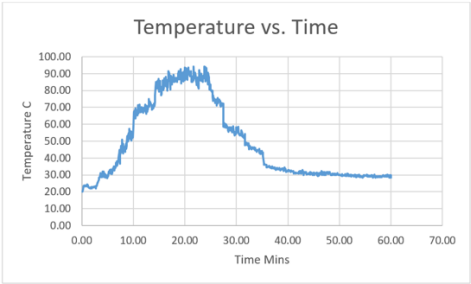
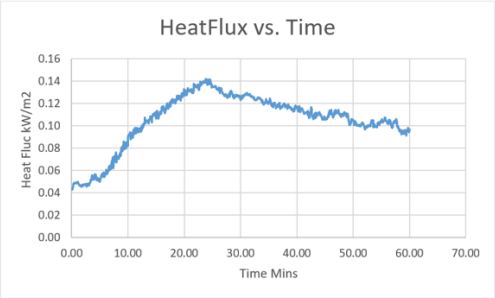
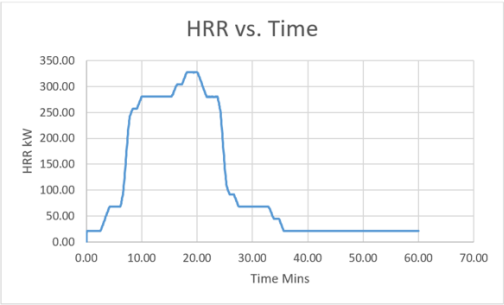


Figure 48. Simulation 9.1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time







Simulations 13, 16, 19 & 22 output:

The Fire Development curves below show a fuel controlled fire with 5,000 MJ of fuel, as demonstrated in Simulations 13, 16, 19, and 22. The energy level increases as more fuel becomes involved in the fire (up to 20 minutes), until all of the available fuel is involved. As the fuel is burned away, the energy level begins to decay (48 minutes to more than 60 minutes). Figures 50 - 59 below show the maximum temperature measured at the thermocouple, the maximum heat flux measured at the heat flux sensor, and the maximum HRR reached for simulations 13, 16, 19, 22.

Figure 49. Simulation 13 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

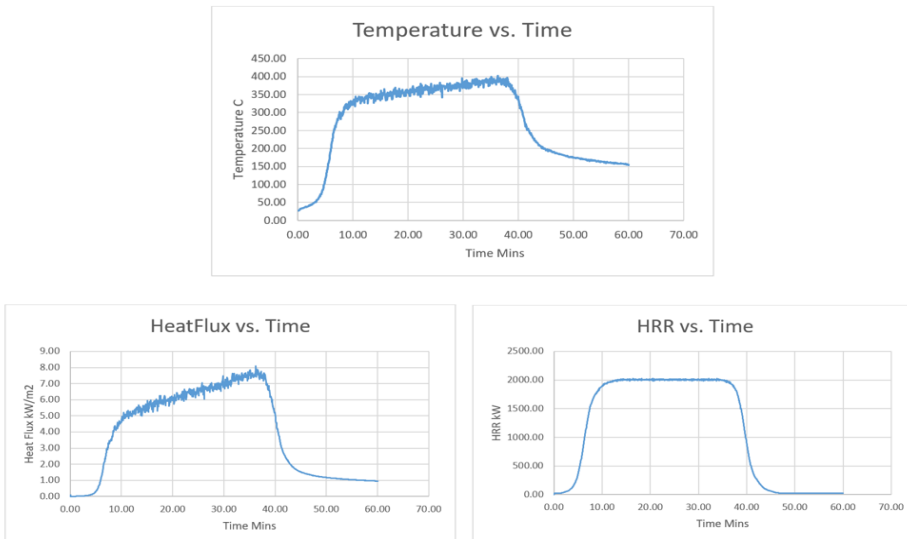


Figure 50. Simulation 13.2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

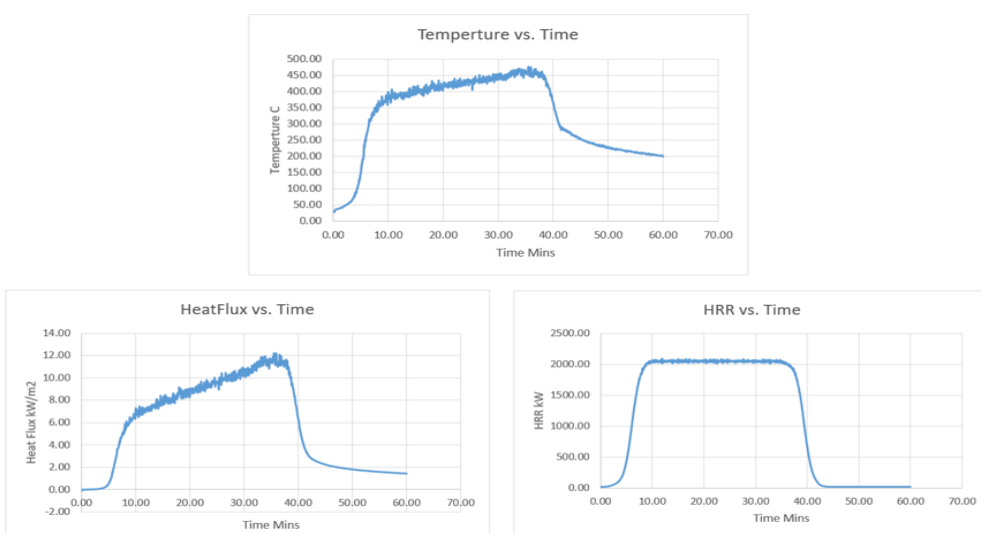


Figure 51. Simulation 16 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

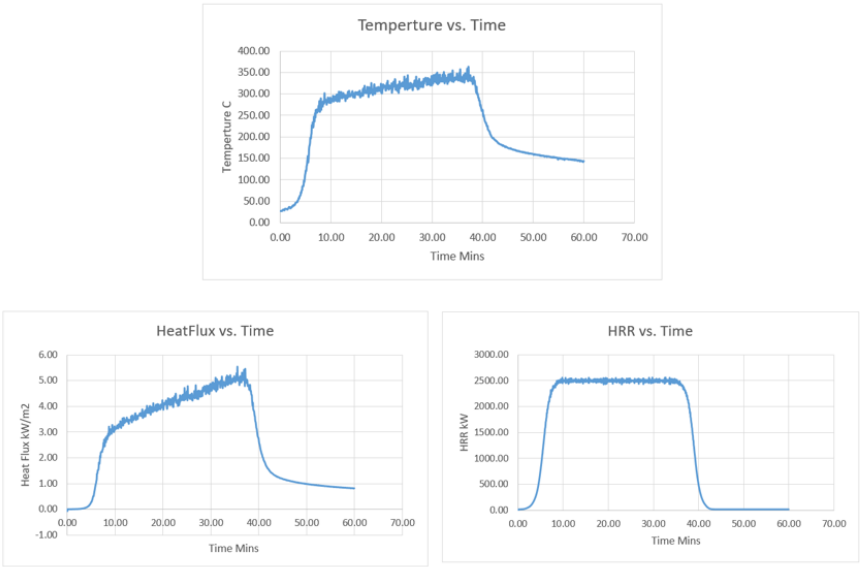


Figure 52. Simulation 16.2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

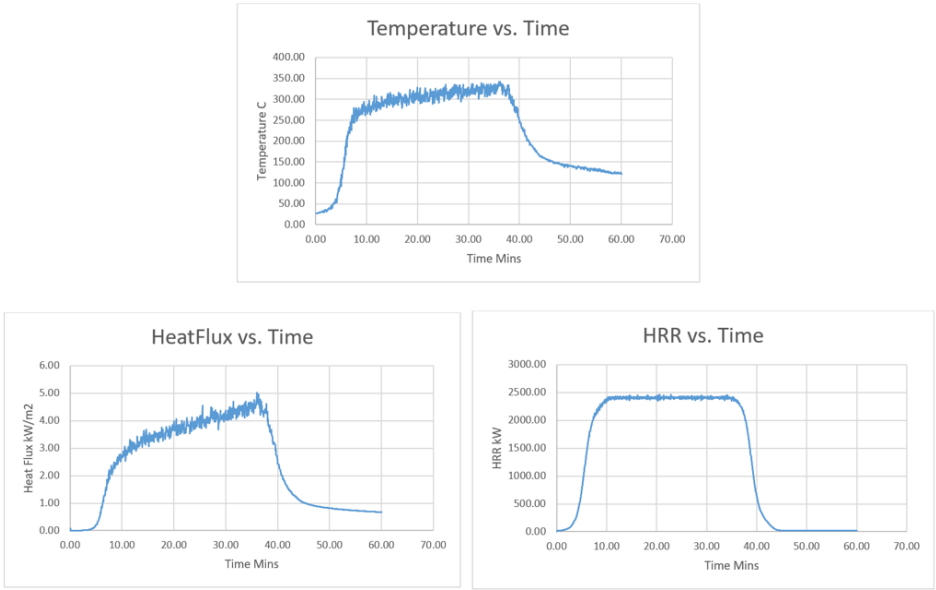


Figure 53. Simulation 19 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

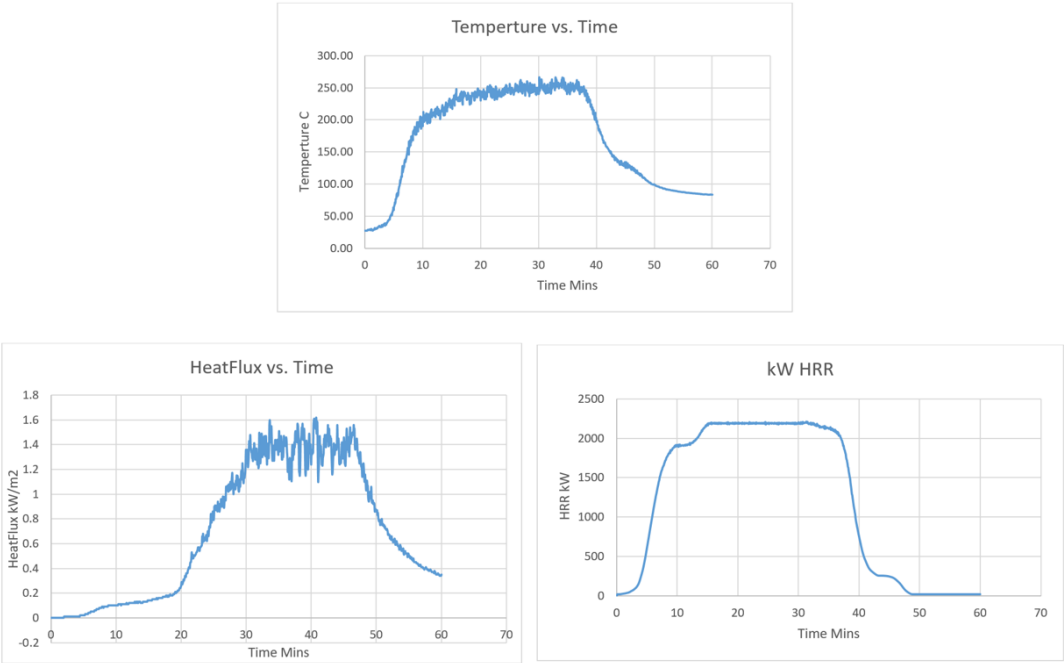


Figure 54. Simulation 19.1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

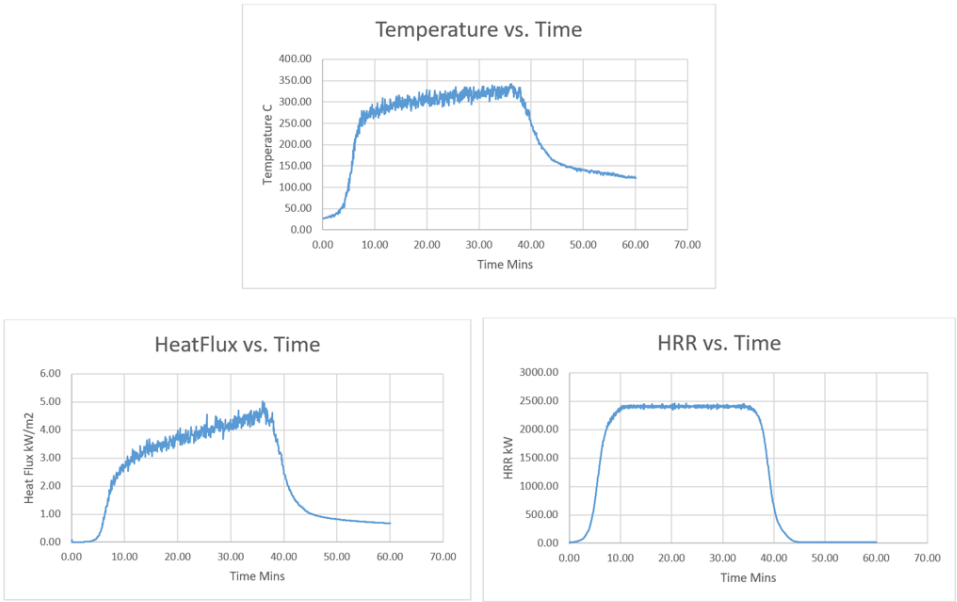


Figure 55. Simulation 19.2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

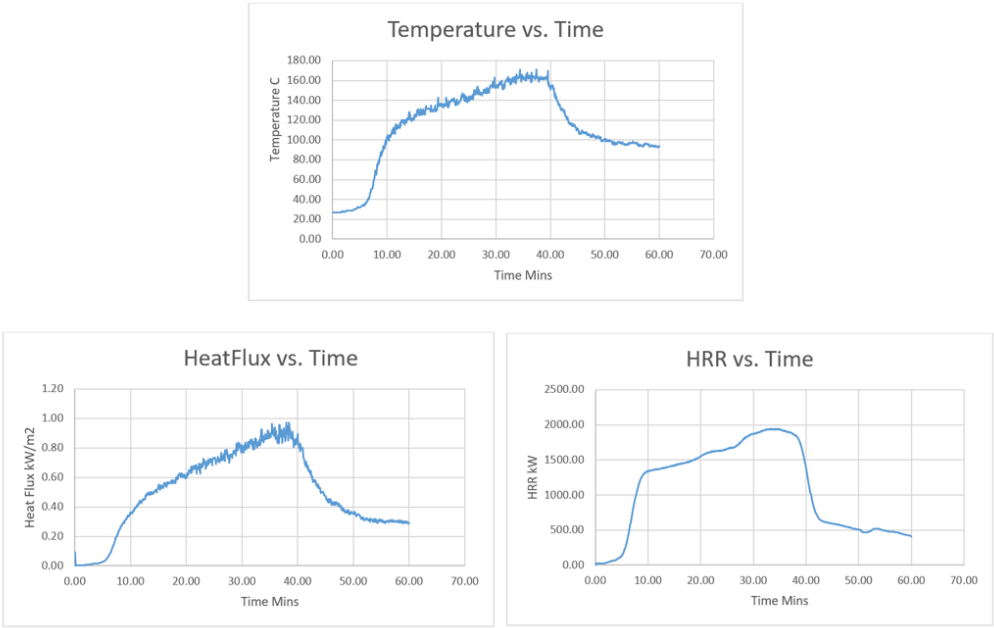


Figure 56. Simulation 22 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

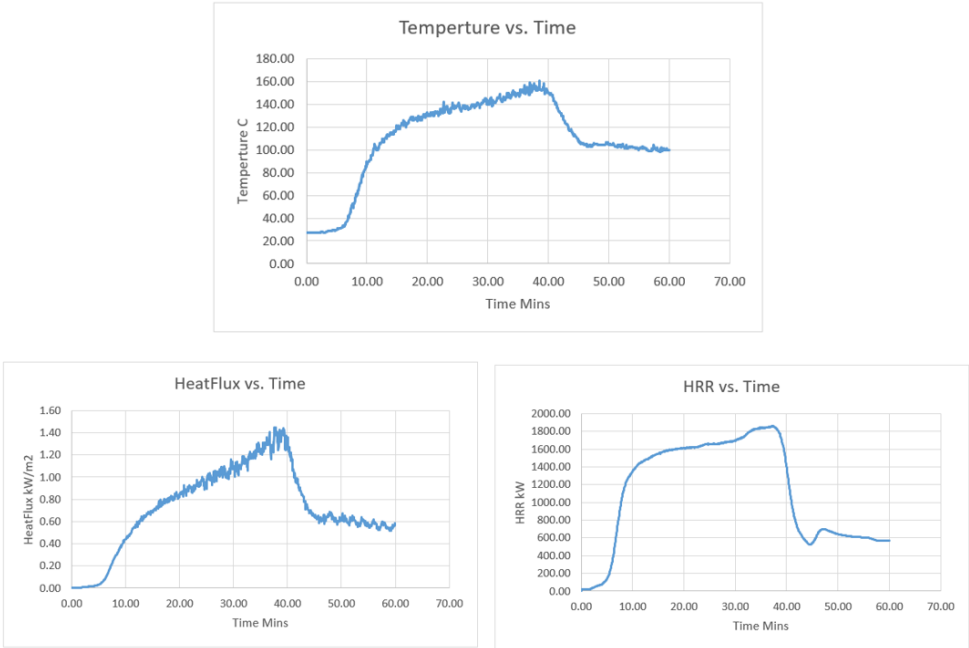


Figure 57. Simulation 22.1 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time

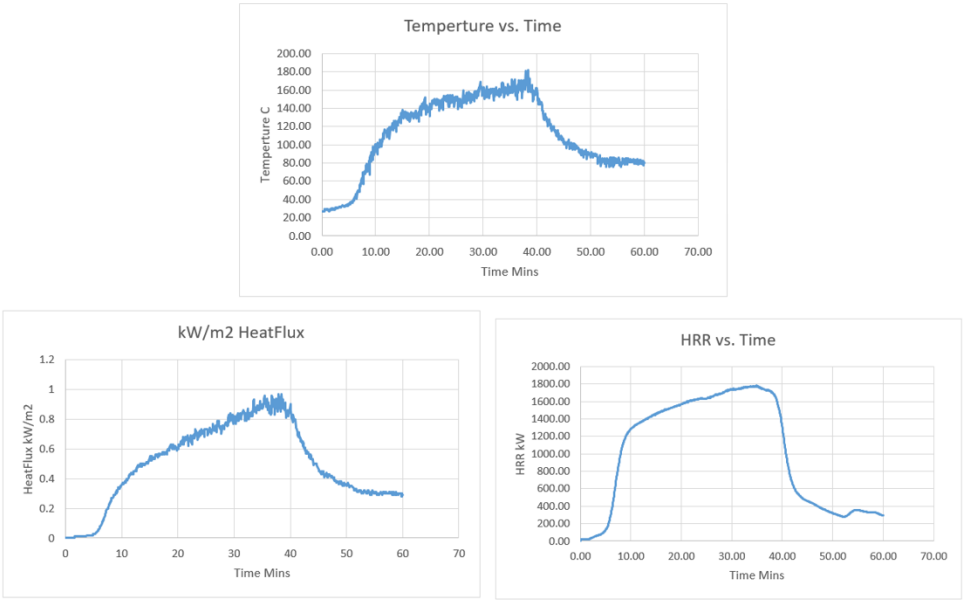


Figure 58. Simulation 22.2 output: Temperature vs Time: Heat Flux vs Time: HRR vs Time