Insights from CANDU Fire Risk Assessment Experience

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Abstract

This technical paper presents the insights gained from the previous CANDU 6 plant fire risk assessments and the fire PSA walkdown of a CANDU 6 plant to support the fire protection considerations of new builds, advanced reactor design, as well as fire PSA projects for other operating CANDU 6 plants. It also discusses the applicable codes and standards, industrial practice and CANDU operating experience.[1]

1. Introduction

For the fire protection program of a nuclear power plant, the regulatory requirements are to (1) prevent significant fires, (2) ensure the reactor safe-shutdown capability, and (3) minimize radioactive releases to the environment in the event of a significant fire.

The approach for fire protection considerations for advanced reactor design is largely rely on deterministic fire hazard assessment and previous experience. A risk-informed integrated approach of using both deterministic and probabilistic assessments for ACR fire protection considerations is not feasible due to lack of a fire PSA model. At the early stage of the project, reactor vendors often developed a simplified design assist Probabilistic Safety Assessment (PSA) model for internal events only. Therefore, feedback and previous fire risk assessment experience from operating plants are valuable insights.

The objective of this paper is to bring insights gained from the previous CANDU 6 plant fire risk assessments and the fire PSA walkdown of a CANDU 6 plant into the fire protection considerations of new builds, thus supplementing the design assist role of the deterministic fire hazard assessment. It also discusses the fire protection philosophy, requirements and industrial practice, as well as insights applicable to the fire PSA modeling and approach for other operating CANDU plants.

2. Fire Protection Considerations and Risk Assessment Approach

The primary objective of the fire protection program in a nuclear power plant is to minimize both the probability of occurrence and the consequences of fire, which are based on a "Defense In Depth" safety concept. Key aspects for an effective fire protection program are to (1) prevent fires from starting, (2) rapidly detect, control, and extinguish fires, and (3) protect systems, structures and components that are important to achieve reactor safe-shutdown should a fire occur.

In addition to fire prevention, detection, alarm and suppression means, the design of the Advanced CANDU Reactor (ACR) follows the functional and physical separation philosophy. Alternative shutdown capability is provided in the Secondary Control Area (SCA) using Shutdown System No.2 (SDS2) that is functionally diverse from SDS1. Redundant divisions are designed for systems that are required for safe shutdown operation, in compliance with fire protection philosophy and separation requirements of Structures, Systems and Components.

[1]. *Canadian Deuterium Uranium (CANDU) operating experience database managed by CANDU Owners Group.*

2.1 Philosophy

The ACR systems are provided with appropriate separation or barriers so that safety functions can be reliably accomplished for postulated fire events. The principles and methods of separation are essentially the same as those implemented in CANDU 6 plants. In addition to the concept of safety system separation (such as between SDS1 and SDS2) used in the past CANDU plants, the ACR separation philosophy also adopts separation between redundant divisions within a system.

2.2 Requirements

The design approach for meeting separation requirements involves the identification and provision of two independent, redundant means of achieving the same safety function. Safety systems are separated from the process systems. In addition, the ACR system design includes independent trains of equipment called divisions. These design requirements ensure that the plant safe shutdown capability is not impaired by a postulated fire event in a limited area.

2.3 Assessment Approaches

Two types of fire risk assessment were conducted for the ACR development project. One is the Fire Hazard Assessment, which is based on a deterministic approach. The other is the Probabilistic Fire Risk Assessment.

(a). Deterministic Assessment

Using a deterministic approach, the ACR Fire Hazard Assessment systematically identify the combustible materials, ignition sources, fire barriers and potential fire growth scenarios, and evaluates the adequacy of fire prevention, detection and suppression means in each pre-defined plant area to ensure the safe shutdown capability.

The ACR Fire Hazard Assessment, in principle, follows the general guidance of Fire Protection for CANDU Nuclear Power Plants stated in CSA N293. It also qualitatively addresses the consequence of fire hazards on plant safety.

(b). Probabilistic Assessment

The Probabilistic Fire Risk Assessment, performed as part of the ACR PSA study, quantitatively address the consequence of fire hazards to plant safety in terms of annual Core Damage Frequency (CDF). This probabilistic assessment derive the fire ignition frequency in each plant area or fire hazard assessment zone, estimate the corresponding probability of fire detection and suppression, develop potential fire propagation models, estimate fire-induced component failure probability, incorporate them into the internal event PSA model, and finally, derive the fire-induced CDF.

3. Industrial Practice

Use of the PSA is encouraged to assist design optimization, system upgrade and back-fitting decisions. However, compliance with the Safety Regulations, Codes & Standards, and General Design Criteria (GDCs) is fundamental. The following three sections discuss the fire protection relevant Codes & Standards, risk-informed alternative for CANDU and LWR nuclear power plants, and Individual Plant Examination of External Events (IPEEE).

3.1 Codes and Standards

The ACR design takes the traditional fire protection codes and standards into consideration. The design of ACR fire protection system fully complies with CSA N293 (Fire Protection for CANDU Nuclear Power Plants), which addresses life safety and business losses. Codes and standards used as guidelines in the ACR design are listed as follows:

NBCC	National Building Code of Canada
ULC 529	Underwriter’s Laboratory of Canada Standard for Smoke Detectors for Fire Alarm Systems
ULC 530	Underwriter’s Laboratory of Canada Standard for Heat Actuated Fire Detectors for Fire Alarm Systems
NFCC	National Fire Code of Canada
NFPA 10	National Fire Protection Association Standard for Portable Fire Extinguishers

NFPA 13	National Fire Protection Association Standard for the Installation of Sprinkler Systems
NFPA 14	National Fire Protection Association Standard for the Installation of Standpipe and Hose Systems
NFPA 15	Standard for Water Spray Fixed Systems for Fire Protection
NFPA 20	National Fire Protection Association Standard for the Installation of Stationary Pumps for Fire Protection
NFPA 25	National Fire Protection Association Standard for the Inspection, Testing and Maintenance of Water Based Fire Protection Systems
NFPA 2001	National Fire Protection Association Standard for Clean Agent Fire Extinguishing Systems
NFPA 24	Standard for the Installation of Private Fire Service Mains and Their Appurtenances
NFPA 804	Standard for Fire Protection for Advanced LWR Nuclear Power Plants

3.2 Risk-informed Alternative for Operating Plants

Light Water Reactor (LWR) licensees in the US are required to have fire protection programs that comply with GDC 3 of Appendix A to 10 CFR Part 50, which requires that Structures, Systems and Components (SSCs) important to safety be designed and located to minimize the probability and effects of fires and explosions.

Similar to NFPA 804 listed above, The GDC 3 is a deterministic based criterion used during design stage. Adoption of NFPA 805 as a risk-informed, performance-based fire protection standard has been approved by USNRC as a voluntary alternative to the existing fire protection requirements at operating plants.

The risk-informed, performance-based approach presents a more realistic prediction of potential fire hazards with a defined approach for quantifying the performance adequacy of fire protection systems. The USNRC considers the NFPA 805 to be an acceptable alternative fire protection program that satisfies the GDC 3 by comparing the following eight elements of an acceptable fire protection program:

1) The delineation of organization, staffing, and responsibilities.
2) A fire hazard analysis sufficient to perform safe shutdown functions and minimize radioactive material releases in the event of a fire.
3) The limitation of damage to structures, systems, and components important to safety so that the capability to safely shut down the reactor is ensured.
4) Evaluation of fire test reports and fire data to ensure they are appropriate and adequate for ensuring compliance with regulatory requirements.
5) Evaluation of compensatory measures for interim use to determine their adequacy and appropriate length of use.
6) Training and qualification of fire protection personnel appropriate for their level of responsibility.
7) Quality assurance.
8) Control of fire protection program changes.

The goals, objectives and performance criteria specified in NFPA 805 maintain the defense-in-depth safety concept of a traditional fire protection program. The standard’s general methodology requires an integrated risk assessment that consists of the acceptability of risk change, defense-in-depth concept and safety margin. The adequacy of the fire protection elements (e.g., control of combustible materials and ignition sources, fire detection, suppression and confinement) can be evaluated using this standard.

NFPA 805 also provides methodologies for post-fire safe shutdown circuit analysis, fire modeling and PSA methods. The performance-based approach of NFPA 805 utilizes the concepts of: establishing a fire damage threshold, using a fire scenario to determine the condition under which a proposed solution is expected to meet the performance criteria, and maintaining sufficient margin between the maximum fire scenario and the limiting fire scenario in the context of protection of required nuclear safety success paths.

3.3 Individual Plant Examination of External Events

Individual Plant Examination of External Events (IPEEE) systematically assesses the plant-specific severe accident vulnerabilities that are initiated by external events. The results of the IPEEE fire analyses provide important insights regarding reactor fire risk. The IPEEE results show that fire events are important contributors to the reported CDF for a majority of plants, ranging on the order of 4 x 10^{-8} to 2 x 10^{-4} core damage events per reactor-year, with the majority of plants reporting a fire CDF in the range of 1 x 10^{-6} to 1 x 10^{-4} core damage events per reactor-year. In some cases, the reported CDF contribution from fire events can exceed that from internal events.

In addition, according to a "Fire Risk Scoping Study" performed in 1989 by Sandia National Laboratories, plant modifications made in response to the new requirements have reduced the core damage frequencies (CDFs) at some plants by a factor of 10. The study also suggested that improper implementation of the regulatory requirements and degradation of fire protection could be risk-significant. The study concluded, for example, that weaknesses in either manual firefighting effectiveness or control system interactions could raise the estimated fire-induced CDF by an order of magnitude.

IPEEEs were conducted for 108 operating units in the US. Three units at two plants (two-unit Quad Cities and Millstone Unit 2) reported fire vulnerabilities in their IPEEE submittals. All others indicated that no fire vulnerabilities were identified based on their IPEEE fire assessment.

Quad Cities initially concluded that fire vulnerabilities existed, but later revised this conclusion based on a re-analysis of the plant. The initially identified vulnerabilities were primarily associated with large oil fires in the turbine building that might compromise the capability of safe shutdown. Factors contributing to this conclusion were the concentration of safety related cables routed through certain areas of the turbine building and the proximity of the remote shutdown panels to significant fire sources. The revised IPEEE fire assessment concluded that the original assessment was excessively conservative, and that the fire-induced CDF was substantially lower than the original estimate. A number of plant improvements were implemented to further reduce the fire-induced CDF, and many of these improvements were credited in the revised assessment. Ultimately, the licensee concluded that there were no fire vulnerabilities at the plant.

Millstone Unit 2 initially identified one fire vulnerability in the IPEEE submittal. A second fire vulnerability was identified during the IPEEE review process. The first fire vulnerability identified by the licensee was associated with the accumulation of significant quantities of transient combustibles (protective clothing) in the vicinity of a large concentration of safety related cables. The second fire vulnerability involved two fire initiators (scenarios) with the potential Conditional Core Damage Probability (CCDP) of 0.1, and was dominated by a loss of the turbine driven auxiliary feedwater system in the turbine building. In contrast, the original assessment concluded that the licensee expected the CCDP for these two scenarios to be 0.002, and the CDF for each scenario was estimated at about 2 x 10^{-8} per reactor-year. A very conservative re-quantification provided in the response to the USNRC questions estimated that the CDF for each scenario is approximately 5 x 10^{-4} per reactor-year. The licensee implemented plant improvements to address these two scenarios. As a result, the core damage frequency for these two scenarios were reduced to approximately 2 x 10^{-8} and 2 x 10^{-7} per reactor-year.

4. Fire Induced CANDU 6 Severe Core Damage Frequency

The following three sections discuss the estimated fire induced Severe Core Damage Frequency (SCDF) from three CANDU 6 Fire PSA projects, as well as their dominant fire hazard areas.

4.1 Generic CANDU 6 Fire PSA

AECL conducted a preliminary probabilistic fire risk assessment of the generic CANDU 6 plant in July 2002 for IAEA to estimate the fire-induced severe core damage frequency and provide insights for generic CANDU 6 design. This assessment was based on the generic fire PSA methodology described in IAEA Safety Report Series No. 10, "Treatment of Internal Fire in Probabilistic Safety Assessment for NPPs", and incorporates previous CANDU experience and engineering judgment.

Fire frequencies in each area were derived based on CANDU fire events and LWR fire events. Fire events were classified in 25 categories, representing one or several of the following characteristics:

- Component Types (such as cables, motors, pumps etc.)
- Areas with Common Characteristics to Other Plants (e.g. MCR)

- Transient Fires (such as welding, transient combustibles, human errors etc.)

Table 1 shows the fire frequency for each of the classified equipment categories of the generic CANDU 6 plant:

Table 1. **Fire Frequency of Equipment Categories (Generic CANDU 6 Plant)**

Category	Category Name	Mean OP Freq./y	Mean SD Freq./y
1	Battery	1.30E-03	1.82E-03
2	Battery charger	2.32E-03	2.43E-03
3	Inverters	1.00E-03	7.36E-04
4	Main control room	3.03E-03	6.16E-03
5	Digital control computers	4.01E-03	6.43E-03
6	Diesel generator sets	2.19E-02	2.24E-02
7	HVAC equipment	3.24E-03	1.84E-03
8	Dryers	5.70E-03	3.97E-03
9	Hydrogen fires	7.46E-03	6.01E-03
10	Logic and protection cabinets	1.85E-02	1.75E-02
11	HTS pumps	3.88E-03	9.94E-03
12	Pumps	1.17E-02	1.08E-02
13	Motor control centre	6.41E-03	7.10E-03
14	Motors	1.07E-02	1.27E-0
15	Motor generator sets	1.34E-03	3.35E-03
16	Power and control cables	1.22E-02	1.58E-02
17	Low voltage switchgear	7.23E-03	5.98E-03
18	High voltage switchgear	1.19E-02	1.48E-02
19	Gas turbine generators	1.11E-02	1.83E-02
20	Turbine generator	2.52E-02	9.01E-03
21	Main unit transformer	1.13E-02	1.11E-02
22	Transformers	1.21E-02	1.03E-02
23	Human error	1.88E-02	2.52E-02
24	Cable fires caused by welding and cutting	1.66E-03	1.01E-03
25	Transient fires caused by welding and cutting	2.92E-02	7.40E-02

Table 2 shows the fire-induced Severe Core Damage Frequency in selected fire zones of the generic CANDU 6 plant:

Table 2. **Summary of Fire Induced Severe Core Damage Frequency (Generic CANDU 6 Plant)**

Fire Zone	Description	SCDF/yr	Percent
FR002	Reactor Building Access Area and Corridors	2.86E-06	11.9%
FR005	Reactor Building Cable Access Way	2.82E-06	11.8%
FR107	Fuelling Machine Vault Room	6.92E-07	2.89%
FR501	HTS Pump Area	1.44E-06	5.99%
Reactor Building Sub-Total		**7.81E-06**	**32.6%**
FS201	Service Building Cable Access Way	1.83E-06	7.62%
S327	Electrical Equipment Room	3.72E-07	1.55%
S328	Electrical Equipment Room	9.16E-07	3.82%
S329	Computer Room	2.77E-07	1.15%
S326	Main Control Room	9.85E-07	4.11%
FS213	Cable Access Area	9.36E-08	0.39%
FS221	ECC HX and D2O Vapour Recovery Area	1.12E-06	4.66%
FS222	Reactor Building Ventilation Exhaust Area	9.64E-07	4.02%
Service Building Sub-Total		**6.55E-06**	**27.3%**
FT001	Turbine Building Cable Access Way	2.76E-06	11.5%
FT111	High voltage Switchgear Room	1.64E-06	6.84%
FT141	Cable Tray Room	6.15E-07	2.57%

FT161	Low Voltage Switchgear and MCC Room	7.08E-07	2.95%
FT174	Inverter Room	3.90E-06	16.2%
Turbine Building Sub-Total		**9.62E-06**	**40.1%**
Total		**2.40E-05**	

The SCDF was estimated assuming the use of non-fire retardant cables. When fire retardant cables were credited, the SCDF was reduced to 1.23E-5/yr. The results show that the fire hazards in the turbine building is greater than that in the reactor building and in the service building. The potential significant fire hazard areas are the inverter room and the cable access areas.

The ignition sources in the inverter room are self-ignited electrical fires and transient fires, which could damage all Group 1 power and control cables in the room. Preliminary estimation showed that it takes about 30 minutes to damage both ODD and EVEN cables. The ignition sources considered in the cable access area are cable fires and transient fires.

4.2 PLGS Fire Assessment

AECL conducted a fire risk baseline assessment of the Point Lepreau Generating Station (PLGS) in December 2003 to estimate the preliminary fire-induced severe core damage frequency. The PLGS assessment report also discussed insights gained from the preliminary probabilistic fire risk assessment of the generic CANDU 6 plant and adopted the same generic fire PSA methodology.

However, the PLGS fire risk baseline assessment yielded different insights as to what are the potential significant fire hazard areas. The analysis concluded that the high voltage switchgear room is the potential significant fire hazard area because the fire ignition frequency of high voltage switchgear is higher compared to other ignition sources, and the cable above may get damaged if a severe fire in the switchgear is not suppressed within 100 minutes.

Table 3 shows the fire-induced Severe Core Damage Frequency in selected fire zones of the PLGS:

Table 3. **Summary of Severe Core Damage Frequency for Each Fire Area (PLGS)**

Fire Zone	**Description**	**SCDF/yr**	**Percent**
R002/3/4	Reactor Building Access Area	7.02E-07	4.55%
R005	Reactor Building Cable Access Area	8.41E-08	0.55%
R012	Fuelling Machine Auxiliary D2O Supply	2.23E-07	1.45%
PHTP	PHT Pump Area	2.82E-07	1.83%
S-146	D2O Vapour Recovery	6.26E-08	0.41%
S-327	Control Equipment Room	4.55E-07	2.95%
S-328	Control Equipment Room	1.87E-06	12.2%
MCR	Main Control Room	7.90E-07	5.13%
T-301	High Voltage Switchgear Room	6.94E-06	45.0%
T-601	600 V Switchgear Room	2.20E-06	14.3%
T-704	Class I&II Equipment Room	1.79E-06	11.6%
	Total	**1.54E-05**	

The SCDF was estimated assuming the use of fire retardant cables and the availability of the steam driven Auxiliary Feeswater Pump (AFW) pump. When the non-fire retardant cables are credited, the SCDF increases to 4.64E-5/yr. When the steam driven AFW pump is unavailable, the SCDF increases to 3.66E-5/yr.

4.3 Wolsong Fire PSA

In September 1997, the Korea Power Engineering Company (KOPEC) conducted a fire risk assessment for Wolsong 2/3/4 Units.

The methodology was based on the Fire-Induced Vulnerability Evaluation (FIVE) approach (TR-100370) and Fire PRA Implementation Guide (TR-105928) developed by Electrical Power Research Institute (EPRI) in April 1992 and December 1995 respectively.

This assessment concluded that the MCR is the most significant potential fire hazard area. This conclusion is quite different than that of the fire risk assessment of the generic CANDU 6 plant and the fire risk baseline assessment of the PLGS.

The fire risk assessment of a generic CANDU 6 plant concluded that the potential significant fire hazard areas are the inverter room and the cable access areas. The fire risk baseline assessment of the PLGS concluded that high voltage switchgear room is the potential significant fire hazard area.

Table 4. **Major Contributions to Fire Induced Severe Core Damage Frequency (Wolsong 2/3/4)**

Fire Zone	Description	SCDF/yr	Percent
SB-009	Main Control Area	2.06E-05	37.1%
TB-001	Gap Between Service Building and Turbine Building	8.20E-06	14.7%
RB-002	Access Area in Reactor Building	7.81E-06	14.0%
RB-010	Fueling Machine Auxiliary Room	4.04E-06	7.30%
SWCB	Switchyard Control Building	3.86E-06	6.90%
XF001	Unit Service Transformer Area	2.99E-06	5.40%
TB014	High Voltage Switchgear Room	1.97E-06	3.50%
TB004	Boiler Feed Pump and Chemical Feed Equipment Room	1.27E-06	2.30%
TB022	Inverter Room	7.42E-07	1.30%
TB055	Condenser Area	7.07E-07	1.30%

4.4 Dominant Fire Hazard Areas

The potential dominant fire hazard areas identified through the fire risk assessment for a generic CANDU 6 plant, PLGS and Wolsong 2/3/4 are different. They are summarized in the following table.

Table 5. **Potential Dominant Fire Hazard Areas of CANDU 6 Plants**

CANDU 6 Plants	Dominant Fire Hazard Areas	Percent	Fire Induced SCDF/yr
Generic CANDU 6 Plant	UPS Inverter Room	16.2%	2.40E-05
Pt. Lepreau Gen. Station	High Voltage Switchgear Room	45.0%	1.54E-05
Wolsong 2/3/4 NPP.	Main Control Room	37.1%	5.56E-05

The high contribution of a MCR fire to the SCDF of the Wolsong 2/3/4 NPP is related to the application of the EPRI generic fire frequency data for electrical panel fires, with equal weighting for each of these panels, and old vintage control cards that are no longer used for the Wolsong 2/3/4 NPP.

The other factor that also influenced the identification of the MCR as a potential significant fire hazard area is that the analysis conservatively considered 15 minutes for fire suppression failure and MCR evacuation. The Sandia test showed only optical density measurements of smoke that exceeded test criteria after 15 minutes. Even with a non-representative smoke intensity, the operator was able to see the panel. Visual measurements from the same test suggested that at least 20 minutes are available before control room panels are visually obscured, and this would reduce the manual non-suppression probability an order of magnitude. Operators will abandon the MCR only as a last resort.

The above comparison confirms that, despite the different dominant fire hazard areas identified in the fire risk assessments of the three CANDU 6 plants, electrical equipment fires have the most likelihood to challenge the safe shutdown of a CANDU 6 plant.

5. CANDU Fire Events in OPEX Database

The operating experience database of the CANDU Owners Group (COG) listed 74 significant fire events that occurred in CANDU plants for the period between first commercial operation to December 31, 1997. Nearly half of the fire events are electrical fires. About 18% could be categorized as human error and transient fires (caused by welding and soldering), 14% categorized as mechanical equipment fires, and 10% categorized as fires affecting the turbine generator. Approximately 5% of fires were related to oil and explosive gases. Insights from this CANDU fire experience confirm that electrical equipment has a relatively higher fire ignition frequency.

In addition, the OPEX database also listed the Narora fire in India (under OPEX Event ID 12689). The event is described as follows:

On March 31, 1993, Narora Unit 1, which was operating at 185 MWe, suffered a catastrophic blade failure, resulting in dynamic imbalance, damage to generator seals, escape, ignition, and burning of hydrogen and oil escaping from ruptured lube oil lines. Cables in the vicinity also caught fire, ultimately resulting in total black out. Core cooling was maintained by thermosyphoning, and make up to boilers was done by diesel operated fire fighting pumps. The fire was completely extinguished after nine hours. The station black out, which included loss of Class I and II power, lasted for a period of 17 hours. The cable fire and lack of proper fire barriers or fire retarding provisions, together with inadequate physical separation in redundant safety related cables, were the main cause of the extended station black out and consequent degradation of several safety systems.

6. PLGS Fire PSA Walkdown

A fire PSA plant walkdown at PLGS was organized by the AECL Pt. Lepreau Refurbishment PSA team from May 23 to June 01, 2004. The walkdown exercise was scheduled during the PLGS maintenance outage time, which provided excellent accessibility to various areas (radiation zone III and zone II) in the reactor building, service building, turbine building and pump house. Cable spreading rooms underneath the MCR and high voltage and low voltage switchgear rooms were not covered in the initial walkdown room list, but were added during the walkdown exercise. A walkdown in the PLGS switchyard was discussed but not conducted during this exercise.

The walkdown exercise did not address how the risk-informed, integrated decision making process should be incorporated into the fire PSA model, and how the fire PSA model can be used to support the optimization of the maintenance activities during the plant outage as part of the entire PSA model.

Major concerns during this walkdown exercise related to the fire PSA are summarized as follows:

- Cable distribution frame in the MCR.
- HTS pump lub. oil leak path to F/M room.
- “ODD” pump cabinet in “EVEN” SCA cabinet room.
- Cable routing involving group one and two systems.
- Incompleteness of cable routine database.

However, the degree of impact on the plant safety in terms of CDF is unknown until a more detailed fire PSA model is established, integrated and evaluated. A brief discussion of the above five items is provided in this paper.

The cable distribution frame in the MCR is a potential fire hazard. The cable distribution frame functions as a large scale wire terminal between incoming cables and the cabinets located in the MCR. It therefore cannot be provided with physical fire barriers. Fire propagation and smoke is a concern, which can affect the MCR habitability and thus impair the plant safety.

On the concrete floor, a gap exists around each of the HTS pump cases. This gap provides a potential lube oil leak path to the F/M room at the lower elevation. The purpose of the gap may be relevant to cope with the mechanical vibration during pump operation. The concern here is that a HTS lube oil fire may propagate to the F/M room through the gap and cause multiple initiating events.

The “ODD” pump MCC cabinet in the “EVEN” SCA room was a construction error. The fire separation in the affected area of the SCA is therefore in question. The SCA was designed later in the project to cope with a seismic event. However, the appearance of an “ODD” pump MCC cabinet in the “EVEN” SCA room needs further study to understand the potential impact on plant safety in a fire event.

The last two items reveal the issues of cable routing design and arrangement. While cable routing of group one and two systems in the same room is deemed to largely affect the effectiveness of the group separation concept in the

CANDU 6 design, the implementation of the “ODD” and “EVEN” separation design concept should be also ensured to reduce the fire-induced severe CDF.

7. Insights for the Advanced CANDU Reactor Project

Insights for advanced reactor fire protection and design improvements, fire risk assessment approach are discussed in this part of the paper.

7.1 Design Insights for ACR

Since the ACR is both an evolutionary and an advanced design that incorporates the previous CANDU design and operating experience, it should also take the future trends into consideration during its preliminary design phase. One of the trends is to use a risk-informed, integrated decision making process. The challenge is how to incorporate the integrated decision making process into the ACR fire protection philosophy and system design.

At this stage of ACR design, a systematic application of the risk-informed, integrated decision making process is technically not feasible for a number of challenges. However, we can perhaps consider the following aspects:

1) While preventing potential damage of multiple divisions, the fire barriers should be arranged as much as possible such that only one fire-induced initiating event could occur.
2) The layout of fire detection and suppression systems should prevent the occurrence of multiple fire-induced initiating events. The layout should also prevent the fire propagation that would lead to the occurrence of the fire-induced initiating event and simultaneous damage of the safe shutdown equipment.
3) Cable layout and separation in the MCR and the spreading room should minimize the fire and smoke propagation and maintain the maximum MCR habitability for safe shutdown.

Other features that could be considered for ACR fire protection are discussed as follows:

(a). Separation of Power Cables

There have been 3 power cable fire events in a CANDU plant from operating experience database. One power cable fire event occurred in a French reactor (it may be related to undersized cable which should really be used for a 5 MW motor, not for 9 MW).

(b). Defense in depth concept in ACR fire protection

The following example demonstrates the defense-in-depth concept in the ACR design. The HTS pressurizer pump room is estimated to have 100L of lube oil for each pump based on the Qinshan CANDU Project. A fire protection program for this room not only reduces the fire frequency in the room, but also prevents a potential HTS pressure boundary failure so that no fire-induced LOCA would occur. Due to civil and layout challenges, the initially proposed 2 hour fire rating of a fully enclosed room with an automatic sprinkler system cannot be implemented. Instead, an oil collection system (similar to the one provided for the HTS pumps) is provided to reduce the fire hazard. Automatic fire detection is provided in the room for timely fire suppression.

(c). Incorporate previous oil spill experience into ACR design and operating manual

The ACR design should feature steel pans to collect leaked or spilled oil and specify in the operating manual the need for the control of oil collection during maintenance activities. These measures would largely minimize the uncertainty of the location of fires.

In addition, the ACR design should minimize the use of hydraulic lube oil for elevators and other equipment as much as possible, prevent cables from fire damage by maximizing the distance to potential fire ignition sources, and have proper cable separation to minimize fire propagation from one division to the other.

Since the ceiling mounted detectors may not be effective for fires started at lower elevations, wall mounted fire detection devices may be installed for high ceiling areas, e.g., beam detectors at half elevation to the ceiling.

7.2 Corrective Actions

The most noticeable design change as a result of an improved fire protection program for the ACR development project is the HTS Pump Motor Oil Collection System. The previous CANDU design does not have such a system to collect the oil leakage from the bearing oil reservoir. Leaked oil provides a potential fire propagation path and could thus affect the plant safe shutdown capability. The fire PSA plant walkdown at PLGS confirmed this potential fire hazard in the operating CANDU 6 plants.

Adding the HTS Pump Motor Oil Collection System for ACR development project provides a solution for an improved fire protection program. This proposed HTS Pump Motor Oil Collection System uses oil collection trays to collect any oil overflow or leakage from the bearing oil reservoir, and directs the oil to a closed tank with a flame-arresting vent or to a temporary oil receiver (such as sumps inside containment). In addition to the oil collection trays, shields are also provided around the pumps to prevent pressurized, leaked oil from dispersing in any direction.

To reduce the electrical cable fire risk in the electrical equipment room next the MCR, the ACR design is no longer adopt the Cable Distribution Frame for the distribution of cable wiring to the electrical and control panels in the electrical equipment room and MCR. The incoming and outgoing cables are connected to individual panels and the distribution of cable wiring is take place inside panels.

7.3 Insights for ACR Fire PSA

The scope and level of detail of the ACR fire PSA are very much dependent on the design information in different stages and the fire protection considerations. A qualitative screening analysis is needed to identify the areas where a fire can trip the plant or affect the safe shutdown equipment. The screening analysis usually involves or interacts with the following:

- Systematic review of the plant layout
- Identification of safe shutdown systems and support systems
- Fire hazard assessment.
- Review of fire protection capabilities

The qualitative screening process can be based on fire zones specified in the fire hazard assessment. Postulated fires in each fire zone can either lead to the occurrence of an initiating event and/or impair the safe shutdown system redundancy. The challenge is that detailed design of the cable routing system is usually not available in the early stage of the project, which makes the fire zone-based screening process rather difficult. A simplified approach that conservatively solves this problem is to adopt the qualitative screening process based on the fire area instead of fire zone.

The fire areas for the qualitative screening process should, in principle, cover all plant areas with special attention to the following buildings:

- Reactor and Reactor Auxiliary Buildings. These buildings contain all safe shutdown equipment and are subdivided by fire barriers into fire areas that correspond to the safety divisions.
- Control Room Complex. This area (including cable spreading room) contains safety-related equipment from all safety divisions in a single fire area.
- Turbine and Switchgear Buildings. These buildings could result in a loss of offsite power and/or a plant shutdown.

After conducting the qualitative screening, a quantitative screening process is needed to screen out the areas where the consequence of a postulated fire, in terms of severe CDF or Large Early Release Frequency (LERF), is below the truncation limit and requires no further detailed analysis.

The numerical values for the quantitative screening analysis can be based on the estimated fire frequency in each area and the Conditional Core Damage Probability derived from the design assist ACR PSA model or the Level 1 full power internal event PSA model. Equipment in the area where a postulated fire occurs is assumed to fail with no credit for the fire suppression system and fire barriers.

A detailed fire propagation model is to be established when more design information is available to identify the ignition sources, combustible materials and the rating of fire barriers. This fire propagation model can then be integrated with the Level 1 full power internal event PSA model and containment event tree to derive the fire-induced severe CDF and LERF.

8. Insights for Shutdown State Fire PSA for CANDU 6 Plant

Increased maintenance activities during plant maintenance outage may introduce additional fire risks. This section discusses how the fire risk assessment during low power and shutdown stage should be dealt with.

Low power and shutdown state operation modes of a CANDU 6 plant are summarized as follows:

MODE 2: Low power, Hot and pressurized (critical or sub-critical)

MODE 3: Cooldown operation (critical or sub-critical)

MODE 4: Very low power, Cold and depressurized (critical or sub-critical)

MODE 5: GSS (hot pressurized, cold depressurized, or drained to header level)

Fire risk in MODE 2 and MODE 3 is judged to be essentially the same as that for the full power operation since most systems that are available at full power continue to be available. Therefore, the fire risk in MODE 2 and MODE 3 can be assumed to be included in the full power fire PSA. The plant will be in these MODES for a relatively short time period.

Similarly, the plant will be in MODE 4 for a short time as well. MODE 5 takes up most of the outage length. Many systems could be in maintenance during this mode, and there are also increased transient fire ignition sources associated with the maintenance activities. Therefore MODE 4 and MODE 5 are subject to the low power and shutdown state fire PSA, which correspond to the following two states:

- Reactor Shutdown, HTS cold de-pressurized and full.
- Reactor Shutdown, HTS cold de-pressurized and drained to headers.

The impact of a fire during shutdown state is significantly different from that during power operation. This difference can be attributed to the following factors:

1) Since the plant is already shutdown and depressurized, failure of shutdown and depressurization is no longer a concern. The only concern related to fire during shutdown state is the ability to continue decay heat removal.
2) Emergency Coolant Injection (ECI) is not available because it is blocked when the reactor is depressurized.
3) Increased maintenance activities and thus a higher probability of fire. However, more personnel may detect and mitigate fire with increased likelihood.
4) Reduced redundancy for decay heat removal systems. System availability varies with the different Plant Operation States.
5) Certain fire barriers may be breached to carry out maintenance activities, which introduce a potential for fire to spread from one division to the other.

Although fire in the Turbine Building may still lead to a loss of offsite power, the magnitude of consequence is judged to be lower than that of full power operation due to an increased time window for operation recovery actions. Similarly, the consequence of a MCR fire is also less significant since the plant is already shutdown and there is sufficient time for the operators to detect and mitigate the fire. If the fire cannot be mitigated, the operator has ample time to control the plant from the SCA.

9. Conclusions

The results of fire assessment conducted for the generic CANDU 6 plant show that fire hazards in the turbine building are greater than those in the reactor building and service building. The potential significant fire hazard areas of the generic CANDU 6 plant are the inverter rooms and the cable access areas. The PLGS fire risk baseline assessment concluded that the high voltage switchgear room is the potential significant fire hazard area. The fire risk assessment conducted for the Wolsong 2/3/4 Units concluded that the MCR is the potential significant fire hazard area.

Despite the different dominant fire hazard areas identified from the fire risk assessments of the three CANDU 6 plants, electrical equipment fire is confirmed to have the most likelihood to challenge the safe shutdown of a CANDU 6 plant. The IPEEE results show that fire events are important contributors to the reported core damage frequency for a majority of plants, ranging on the order of 4 x 10^{-8} to 2 x 10^{-4} core damage events per reactor-year, with the majority of plants reporting a fire CDF in the range of 1 x 10^{-6} to 1 x 10^{-4} core damage events per reactor-year. In some cases, the reported CDF contribution from fire events can exceed that from internal events.

Based on the insights gained from the fire PSA walkdown of a CANDU 6 plant, corrective actions have been taken in the ACR design. The fire hazard in the F/M room resulted from the leaked oil of the HTS pump motor is eliminated by adding the HTS Pump Motor Oil Collection System in the ACR design. In addition to the oil collection tray, shields are also provided around the HTS pumps to prevent pressurized, leaked oil from dispersing in any direction. The cable distribution frame is not adopted in the ACR design for the distribution of cable wiring to the electrical and control panels in the electrical equipment room and MCR. The ACR design also adopts the division separation concept in addition to the traditional safety system separation concept used in the CANDU plant design. All these measures can contribute to a reduced fire-induced severe CDF of the ACR.

10. Limitations

The ACR is an evolutionary design of CANDU reactors. Plant layout, system configuration and separation may be significantly different than the previous CANDU 6 design concept. The ACR design is also intended to meet both Canadian and US regulatory requirements. As such, insights from previous CANDU 6 fire assessment experience may not be directly applicable to the ACR project, especially when the detailed layout of electrical equipment and cable routing is not available upon.

It should also be noted that all of the probabilistic fire risk assessments previously conducted for CANDU 6 plants as discussed in this paper only address the fire-induced severe core damage frequency over a 24 hour mission time period. This time period is intended for achieving hot shutdown and decay heat removal without repair activities, though cold shutdown is normally achieved within 6 hours. If equipment required for achieving cold shutdown is damaged by fire, the ability of achieving cold shutdown is expected to be restored within 72 hours, and repair activities are allowed to be credited in the risk model. In the regulatory context, the fire safe-shutdown analysis should consider the ability to achieve and maintain cold shutdown.

References

[1] AECL, "Generic CANDU Probabilistic Safety Assessment –Reference Analysis", Rev. 0, July 2002.

[2] "A Summary of AECL Review of Wolsong PSA by KOPEC and KAERI", April 1998

[3] EPRI, TR-100370, "Fire-Induced Vulnerability Evaluation (FIVE)", April, 1992.

[4] EPRI, TR-105928, "Fire PRA Implementation Guide", December 1995.

[5] NUREG/CR-1742, "Perspectives Gained From the Individual Plant Examination of External Events Program," U.S. Nuclear Regulatory Commission, Washington, DC, April 2002.

[6] NUREG/CR-5088, "Fire Risk Scoping Study: Investigation of NPP Fire Risk, Including Previously Unaddressed Issues", January 1989.

[7] NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities", October 2004.