ANALYSING SUPERCRITICAL WATER REACTOR'S (SCWR'S) SPECIAL SAFETY SYSTEMS USING PROBABILISTIC TOOLS

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Abstract

The next generation of reactors, termed Generation IV, has very attractive features – its superior safety characteristics, high thermal efficiency, and fuel cycle sustainability. A key element of the Generation IV designs is the improvement in safety, which in turn requires improvements in safety system performance and reliability, as well as a reduction in initiating event frequencies. This study compares the response of the systems important to safety in the CANDU-Supercritical Water Reactor to those of the generic CANDU under a main steamline break accident and loss of forced circulation events – to quantify the improvements in safety for the pre-conceptual CANDU SCWR design. Probabilistic safety analysis is the tool used in this study to test the behavior of the pre-conceptual design during these events.

1. Introduction

The Supercritical Water Reactor (SCWR) is a Generation-IV (GEN-IV) reactor, and as such is expected to fulfill the GEN-IV aim of having advances in fuel cycle sustainability, safety, performance, proliferation resistance and physical protection. Canada's focus within the Generation IV context is on the SCWR design due to its high thermal efficiencies, potential for improved safety, and because it is a proven technology in the fossil power plant sector. However, the CANDU-SCWR will differ greatly from the traditional CANDU in fuel channel design, core orientation, fuel, and coolant type. Some similarities remain, including it being a pressure tube design and heavy water-moderated. The comparisons between the reactors are shown in Table 1 below.

Table 1: Comparing Generic CANDU to SCWR operating conditions

Parameter	**CANDU**	**SCWR**
Inlet Temp	290°C	350°C
Outlet Temp	320°C	625°C
Outlet Pressure	10MPa	25MPa
Core orientation	Horizontal	Vertical
Fuel	Natural uranium	Enriched uranium or thorium
Coolant	Heavy water	Light water
Moderator	Heavy water	Heavy water

Another key difference in these two reactors is that the SCWR will use a direct cycle similar to the BWR and ESBWR designs. Therefore there will be no steam generators as the supercritical fluid from

the core will go directly to the turbine-generator. The direct cycle combined with the higher operating conditions are expected to produce an increased thermodynamic efficiency of over 45% [1], compared to the just over 30% of the CANDU, as well as greatly simplify the design of the secondary side system.
The higher temperatures and pressures in the SCWR design pose unique materials selection issues such as corrosion [2] for in-core components, as well as impacting the excepted frequency of some abnormal or accident events. This work examines the pre-conceptual CANDU-SCWR design using PRA methods to quantify the level of improvements in risk. The SCWR system proposes to improve the risks associated with design basis accidents by improving component reliability, including passive safety features, simplification of the design, and improvements in safety system reliability. The design also proposes to reduce the probability and consequences of beyond design basis and severe accidents by adopting passive cooling features for events such as station blackout. The aim of the work is to compare the safety systems' response to postulated Design Basis Accidents or Anticipated Operational Occurrences.

Probabilistic safety analyses tools are used in this study to test the behavior of the pre-conceptual design during accidents. The paper presents the preliminary findings, reliability metrics and projected Core Damage Frequency. These results will be compared to the traditional CANDU's response in similar events to quantify the improvements to safety from this GEN-IV design.

2. SCWR improved safety through passive systems

As a GEN-IV reactor, the SCWR is intended for superior safety by incorporating additional passive safety concepts. Passive safety systems are desirable because they greatly improve the reliability of systems under adverse conditions and accidents. 'Passive systems' are defined in this work as systems which can meet their design requirements without the need for external power (either electrical or mechanical). Therefore, passive systems are expected to have higher reliability and functionality since loss of electrical and motive power will not hinder their ability to mitigate or prevent certain accidents or sequences. Passive systems greatly simplify plant systems and equipment, as well as potentially improve maintenance related issues (i.e. less moving parts and mechanical equipment to test and maintain). Typical passive safety systems include natural circulation based heat sinks, gravity driven feed systems, and negative power coefficients. These passive systems can be combined with proven active safety systems to optimize the plant risk.

One of the ways the SCWR can incorporate passive safety concepts is to have a tank of water in containment that can act as both a supply/make-up water tank for the heat transport system, and a heat sink in accident scenarios. (One of the accidents analyzed here will assume such a tank is present in the SCWR design.) The water will be at atmospheric pressure and (containment) room temperature. This tank will need to supply water or be a heat sink passively so as not to introduce a great source of unreliability. For instance, it can operate on pressure difference to open valves that release water. The In-containment Refueling Water Storage Tank (IRWST) can have a heat exchanger included to improve the decay heat removal capability and lengthen the duration of passive heat removal. So when the IRWST is required, the core transfers heat to the coolant which becomes less dense and goes up to the IRWST. There, the coolant becomes cooler and less dense and falls (by gravity) back to the core. Thus, a natural thermosyphoning path is established. The reliability or stability of thermosyphoning leading to a safe outcome is dependent on the eventual SCWR design. It is not yet known the duration that this process will be used as a heat transfer option; thermosyphoning depends on the temperature gradient present, which depends on the final design of the reactor. Even if the IRWST is available, the

SCWR is being designed to have a moderator that can remove decay heat under accident situations without the need for the pressure tubes to fail – as is the case in CANDU-type reactors. The Moderator Cooling System (MCS) will be a passive system, thus increasing its reliability. It can therefore be a backup heat sink for accidents in the SCWR.

It should be noted that passive safety systems can never be 100% reliable. Passive safety systems produce challenges of their own in terms of testing them to ensure they have the required performance reliability. Passive systems also may fail to meet their design objectives due to aging of the equipment, thermal shock, mechanical damage and/or external events. Therefore, passive systems should be designed, maintained, and tested to ensure they retain their expected high reliability.

3. Probability basis

Probability theory is widely used in the field of reliability engineering to analyse the risk associated with a system, or to determine the reliability of a component. Various components and systems have been so analysed and it is now possible to know their failure rate based on the probability of failure.

There are different kinds of failures. **Demand Failures** arise in equipment that fail to start (or fail to open) e.g. Auxiliary Feedwater, Shutdown Cooling, Emergency Power. These are represented generally by the Binomial Distribution:

$$P\{r\ failures\ in\ N\ trials\ |p\} = \binom{N}{r} p^r (1-p)^{N-r} \tag{1}$$

where for M-out-of-N combinations:

$$\binom{N}{r} = \frac{N!}{(N-r)!\,r!} \tag{2}$$

P = Probability of failure for a single demand, and p = number of failures/no. of demands
Eqn. (2) can be used to convert frequencies of failures to probabilities.
Another failure type is the **Run Failures** [3]. This is for equipment that fails to run after a certain time, for example a pump is run from 0 to 100hrs and then fails. So these failures are time-related or time-dependent. Examples of continuous systems are the reactor power control system, charging systems and air coolers. They are represented by the Poisson distribution:

$$P\{r\ failures\ in\ (0,t)|\lambda\} = \frac{(\lambda t)^r e^{-\lambda t}}{r!} \tag{3}$$

where λ = probability of failure
The probability of one or more failures simplifies to exponential:

$$P\{T_f < t \mid \lambda\} = 1 - e^{-\lambda t} \approx \lambda t \qquad (for\ small\ \lambda t; when\ \lambda t < 0.1) \tag{4}$$

For a series of independent events, the probability of them all occurring can be approximated by the sum of the individual probabilities of them occurring.

That is, for a series of events E_1, E_2, … E_n, the probability of them all occurring is:

$$P(E) = P(E_1) + P(E_2) + \ldots + P(E_n) - \prod_{i=1}^{n} P(E_i) \tag{5}$$

But for small probability values of the events, the last term can be neglected [4], approximating to

$$P(E) = P(E_1) + P(E_2) + \ldots + P(E_n) \tag{6}$$

3.1 Probabilistic assessments

A Probabilistic Risk Assessment (PRA) is a tool to identify and analyse accident scenarios, and to estimate the likelihood and consequences of each accident. A PRA is beneficial because its results entail a comprehensive look at the plant under investigation, like the plant design, its operational practice and history, and the reliability of the plant's systems and components. So the PRA method uses an in-depth examination to generate as many accident scenarios as possible. Then the list is culled to events with a significant risk value, where "risk" can be evaluated from eqn. (7) below:

$$Risk = Expected\ Frequency\ of\ undesired\ event\ \times Expected\ Consequence \qquad (7)$$

Some advantages of the PRA are that it gives the analyst a great deal of insight into the design, it allows one to see the complex interactions and dependencies in the design, and allows one to perform sensitivity and uncertainty analysis. Thus, the analyst can detect effects of different inputs so different scenarios are discovered. The PRA also allows one to see what elements have a major impact on the plant's risk profile: e.g. does it expose workers more, or increase the frequency of severe core damage? Some of the drawbacks of the PRA are: (a) Lack of understanding of certain processes (e.g. depressurization profile of HTS after LBLOCA) and how to model them. (b) Some models require significant assumptions. Therefore, design uses inputs from probabilistic and deterministic approaches to perform risk analysis. This study's application of PRA tools to the SCWR design is preliminary since a) at this time not all systems and components for the design have been identified, b) some proposed materials and equipment do not yet exist, and there are no operating experiences to draw from for many SCWR systems.

3.2 Probabilistic analysis methods

A PRA can be conducted with the use of event trees and fault trees. This study used these tools from the CAFTA computer program. Event Trees are used to model the safety functional response that mitigates an initiating event. Some of the steps taken in constructing the event trees include:

1. Identifying initiating event
2. Determining systems available to perform the safety functions
3. Determining what constitutes "success" for the systems required for the plant safety functions
4. Constructing the tree
5. Determining the frequency of the event and the probabilities of the event tree's branches
6. Calculating the probabilities for the identified consequences

When constructing event trees in this study, most events are deemed independent and so the final probability outcome is obtained by summing the frequencies of the events using eqn. (6).

Fault Trees enable one to define all the credible ways in which an undesired event can occur. It helps to show which systems are dependent on others. The uses of the fault tree include:

1. Visual display of model of the physical system
2. Helps identify weaknesses or vulnerabilities in the system
3. Identify interrelationships between events
4. Can help eliminate costly design changes

The CANDU-SCWR is in the pre-conceptual stage of design and as such, there is no model from which to draw dimensions, figures, or operating records. The responses of various components and systems would have been used in determining failure rates and probabilities. Therefore, to get values to use in the SCWR's fault trees and event trees, the generic CANDU (and in some cases LWR) failure probabilities were used based on a variety of sources: NUREG/CR-2300[5] and NUREG/CR-75/014 (WASH 1400) [6], IAEA-TECDOC [7] and the SRS Generic Database [8].

For the equipment which would have run failures, their probability of failure was calculated using eqn. (4), since λ is given in these documents usually by the daily or hourly rate. Also, it was assumed that the reactor was shut down for 28 days every 2 years, therefore a capacity factor of 0.96 was applied as an adjustment on the failure rate of each component taken from ref. [7]. For instance the probability of failure of a diesel generator is given as 1.6E-02. When multiplied by 0.96, the probability of the generator failing to start is 0.01536/yr.

4. Accident analysis

Two accidents were chosen to examine the response of the SCWR's safety systems: a Steam Main Break accident, and a Loss of Forced Circulation. The latter was brought about by a loss of Class-IV power. When an accident occurs, the systems responsible for controlling or stopping the accident are either initialized automatically or a signal is given to the operator so that the operator can manually stop the process. The Event Trees will show both automatic systems working, such as the shutdown systems, and the manual operations such as initiating cooling with the shutdown cooling system.

The salient systems considered as part of the SCWR design are:

i. A heavy water moderated and light water cooled vertical core design with pressure tubes using the AECL-proposed High Efficiency Channel (HEC) design. The flow orientation is upwards from a lower to upper plenum to promote natural circulation in the event of loss of forced flow.

ii. Direct cycle heat transport system with similar ex-core components and characteristics as a BWR or ESBWR, with the exceptions being materials, pipes and turbines being selected for super critical water pressures and temperatures.

iii. Two special safety systems with at least one effective trip parameter for each design basis accident. Both shutdown systems (SDSs) are of diverse design, and are physically and functionally separate from each other, systems, and from other special safety systems. Either of the SDSs is capable of shutting down the reactor completely. Both shutdown systems are fast-acting; it is assumed the SCWR will use the similar SDSs mechanisms as existing CANDUs with the exceptions being a 2-out-of-4 logic system with enhanced component reliabilities.

iv. An Automatic Depressurization System (ADS) and Emergency Coolant Injection (ECI) safety system, The ADS facilitates blowdown cooling and coolant recovery as well as system depressurization which subsequently enables a switchover to either the low power cooling system (see below) or a natural circulation pathway. The ADS is activated by opening valves connecting the system to the main heat transport system (HTS) and provides a pathway for blowdown cooling flows to a suppression pool. Water from this pool is then either gravity fed or fed by the ECI system back into the HTS under low pressure and temperature conditions.

v. A containment safety system similar in nature to existing CANDU designs which prevents and controls active material releases. This containment is designed to withstand the pressure, temperature, pipe whip and turbine disassembly forces for design basis events and remain structurally intact.

vi. A low-power and decay heat removal system similar in nature to the existing CANDU reactor shutdown cooling system which can be used for start-up conditions, maintenance and as an emergency heat sink. The system is capable of cooling the reactor up to 3%FP at pressures and temperatures relevant to the SCWR system up to the point where the main heat transport system pumps can accommodate the process.

vii. A passive driven moderator coolant system which also acts as an ultimate emergency heat sink. In any loss of heat sink event, the fuel and pressure tube components have been selected to allow for heat transfer directly after reactor trip and under decay heat loads to the moderator without any active equipment.

4.1 Loss of forced circulation event

There are four classes of power supplies for the CANDU. Rated according to reliability, Class I is highest and Class-IV is lowest. Class IV power can be obtained in CANDU systems either directly from the electrical grid or from station generators inside the plant (or both). While class IV power is necessary for reactor operation at high power, interruption of Class-IV power should have no direct effect on reactor safety.

Class-III power is used for critical system loads such as low power cooling, auxiliary feedwater, station instrument air, as well as to provide power to Class II and Class I systems. Class III is supplied either from the Class IV systems or from diesel generators if Class IV is unavailable. Class-II power is the uninterruptible source of AC power for emergency lighting, station computers, and essential motor loads, while Class-I power is the uninterruptible DC power supply for essential equipment such as DC motors, reactor safety equipment, and control logic.

Loss of Class-IV power event can be brought on by a station disconnection from the grid or loss of offsite power coupled with loss of station-generated power. The event can also stem from a turbine or reactor trip and a failure of the electrical system to switch to grid supplied power. The Class-IV power system supplies all the large loads directly e.g. motors for heat transport pumps, condensing cooling water, and other service water pumps. Under normal operating conditions, all other modes of power are supplied by Class-IV. Loss of Class-IV power means the plant can no longer produce power; if sustained, the reactor will have to be shut down. Even with the loss of Class-IV power, the pressure, flow, and power signals will be powered by the uninterruptible Class- I and -II power supplies. A Loss of Class-IV power and a failure of the Class III power supplies will lead to a station-blackout and subsequent loss of heat sink accident, which must be considered during design and safety analysis.

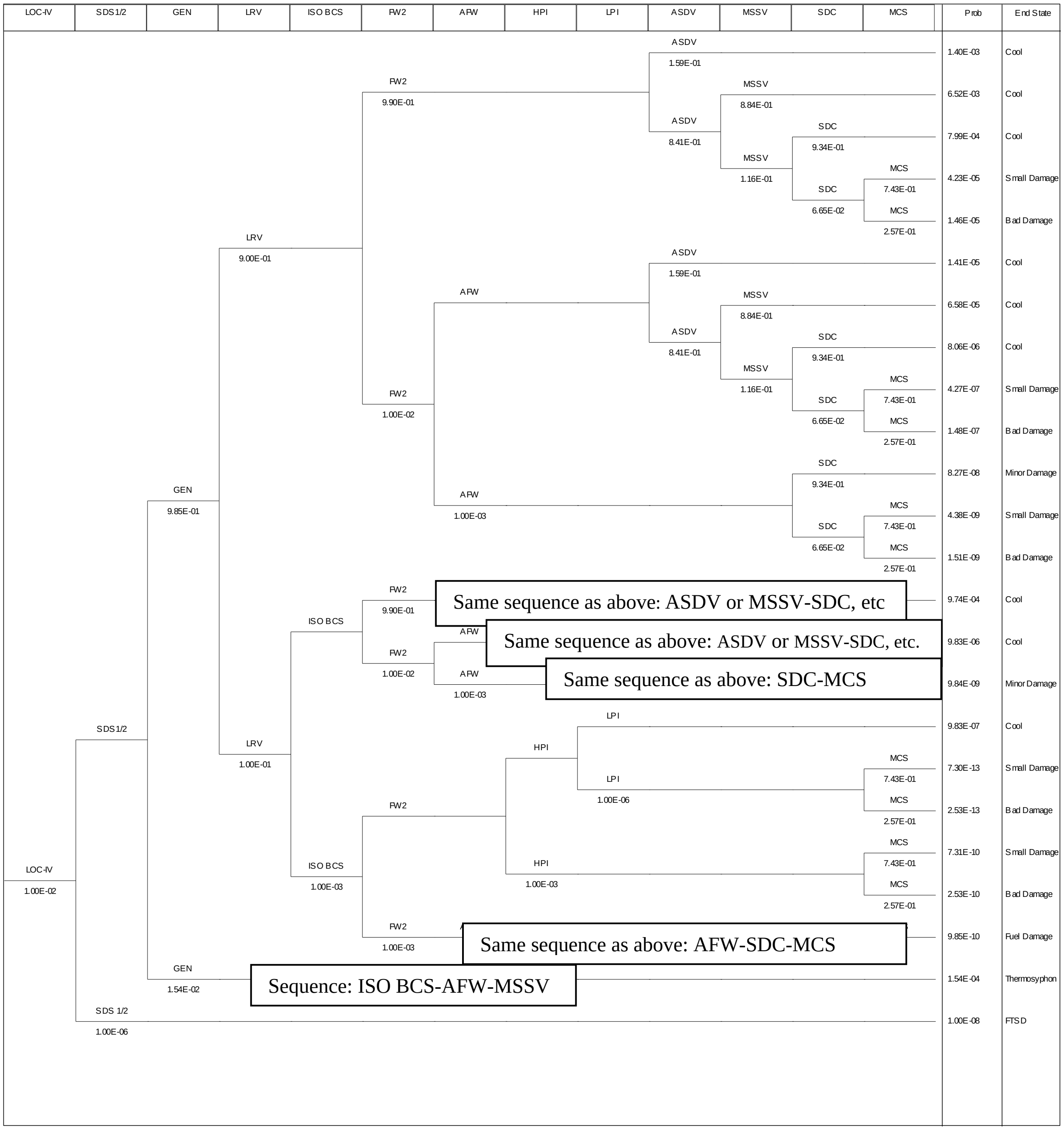
4.1.1 LOSS OF CLASS-IV POWER – CANDU

One of the internal events that initiate Loss of Class-IV power is a turbine trip. Table 2 is an event tree showing a possible sequence for the progression of this accident, for which the description is below:

1. On turbine trip, the reactor regulating system initiates a stepback because the steam generator pressure increases due to the reduction of flow to the turbine. This will lead to the reactor shutting down. E*vent: Shutdown systems (SDS1/2)*
2. The heat transport pumps are run down, removing decay heat in the process. This takes 2-3 minutes and ensures a smooth transition to thermosyphoning. Then, the diesel generators should start, supplying Class-III power. Hence, the SDCS will pick up the long term heat sink function. Without the Class-III generators, there will be no SDCS pumps, nor Emergency Core Cooling (ECC) pumps, boiler feedpumps, nor heat transport feedpumps. *Event: Generators (GEN)*
3. To avoid over-pressurization of the HTS, the liquid relief valves (LRV's) may open periodically during the transient to limit pressure. *Event: LRV*

4. If the generators start, thermosyphoning can stop and the SDCS can be used to cool the core coupled with heat sinks either in the steam generators with auxiliary feedwater or through the SDCS heat exchangers. Group1 feedwater system will be unavailable with Loss of Class-IV, so Group 2 feedwater can be used (or backup Auxiliary Feedwater System). *Event: FW2, AFW*
5. If the LRV's refuse to reclose, the inventory will keep going to the Bleed Condenser (BCS), to the D_2O storage tank, and into the reactor building. The bleed condenser can be isolated so that HTS inventory is not lost to D_2O tank. Failure to isolate the BCS and will cause an event similar to a loss of primary coolant. *Event: Isolate BCS (ISO BCS)*
6. Crash Cooling is used if the generators come on yet both LRV & BCS fail. Crash cooling of the steam generator quickly removes heat from the reactor core. Crash Cooling cools the coolant in the steam generator and allows HTS pressure to fall enough to initiate ECC. The Atmospheric Steam Discharge Valves (ASDV) or MSSV can be used. *Event: ASDV, MSSV*. If LRV and BCS failed, Crash Cooling and ECC needs to activated. That means using High Pressure Injection (HPI) and Low Pressure Injection (LPI) to cool core. *Event: HPI, LPI*
7. In the unlikely event that the MSSV's fail, use Shutdown Cooling System (SDCS) to remove decay heat. *Event: SDC*
8. The Moderator Cooling System (MCS) acts as a heat sink if all other heat sinks have failed. In this event pressure tubes either sag or balloon into contact with the calandria tubes and heat is conducted to the moderator system. Some pressure tube and fuel overheating may occur during this event. *Event: MCS*

Table 2: Event Tree showing Loss of Class-IV Power accident in CANDU



4.1.2 LOSS OF CLASS-IV POWER – SCWR

The SCWR sequence will deviate from the generic CANDU sequence due to the direct cycle nature of the system. Upon turbine trip and loss of Class-IV power, the shutdown systems will need to activate. At this point the ADS system should valve in and allow blowdown cooling, and then ECI (alternatively named Emergency Feedwater System, EFWS) through the suppression pools, which is similar to the ESBWR ADS system design. Thus the initial blowdown cooling is passive after the valve actuation of the ADS. After a period of time the pressure in the HTS will have decreased sufficiently, as well as an appreciable decrease in decay heat, to allow for the non-safety low power cooling system to be initiated. If this process system is unavailable, the low pressure ECI system is used to circulate fluid from the suppression pool to the core. In the long term, liquid in the suppression pool will be cooled using a passive heat exchanger system to the environment. In the event of a failure of ADS and/or ECI and/or low power cooling systems modes, the heat sink may be maintained by the purely passive moderator system operating in its natural circulation mode.

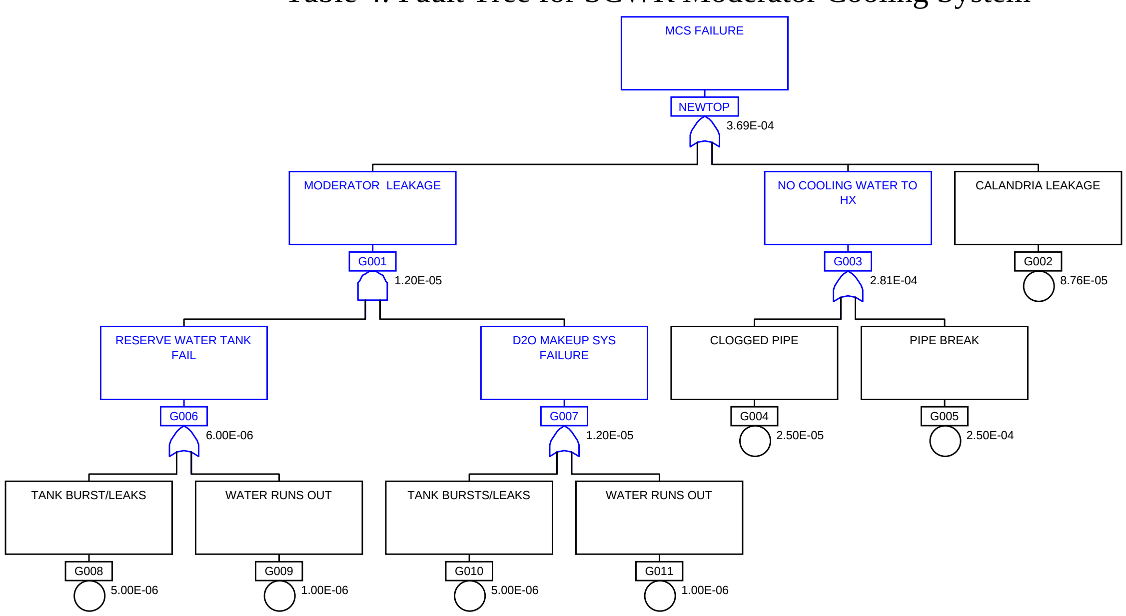
The difference in outcomes for the state of the core following the accident is shown in Table 3. The probability of failure for the events is listed.

Table 3: Probability of failures from Loss of Class-IV power accident

OUTCOME	CANDU	SCWR
Cool core	9.84E-03/yr	9.94E-03/yr
Small damage	4.73E-05/yr	3.63E-17/yr
Bad damage	1.63E-05/yr	3.05E-06/yr

Where ‘Small damage’ implies pressure tube is twisted and destroyed in some places; release of fission product with damage to cladding; fuel bundle integrity might remain. ‘Bad damage’ implies extensive fuel damage; large release of fission product also.
For systems comprising various components without generic failure rates, the system’s overall failure rate was derived from the Fault Trees constructed. For instance, the reliability of the SCWR’s MCS was calculated from the Fault Tree in Table 4, giving 3.69E-04/yr.

Table 4: Fault Tree for SCWR Moderator Cooling System



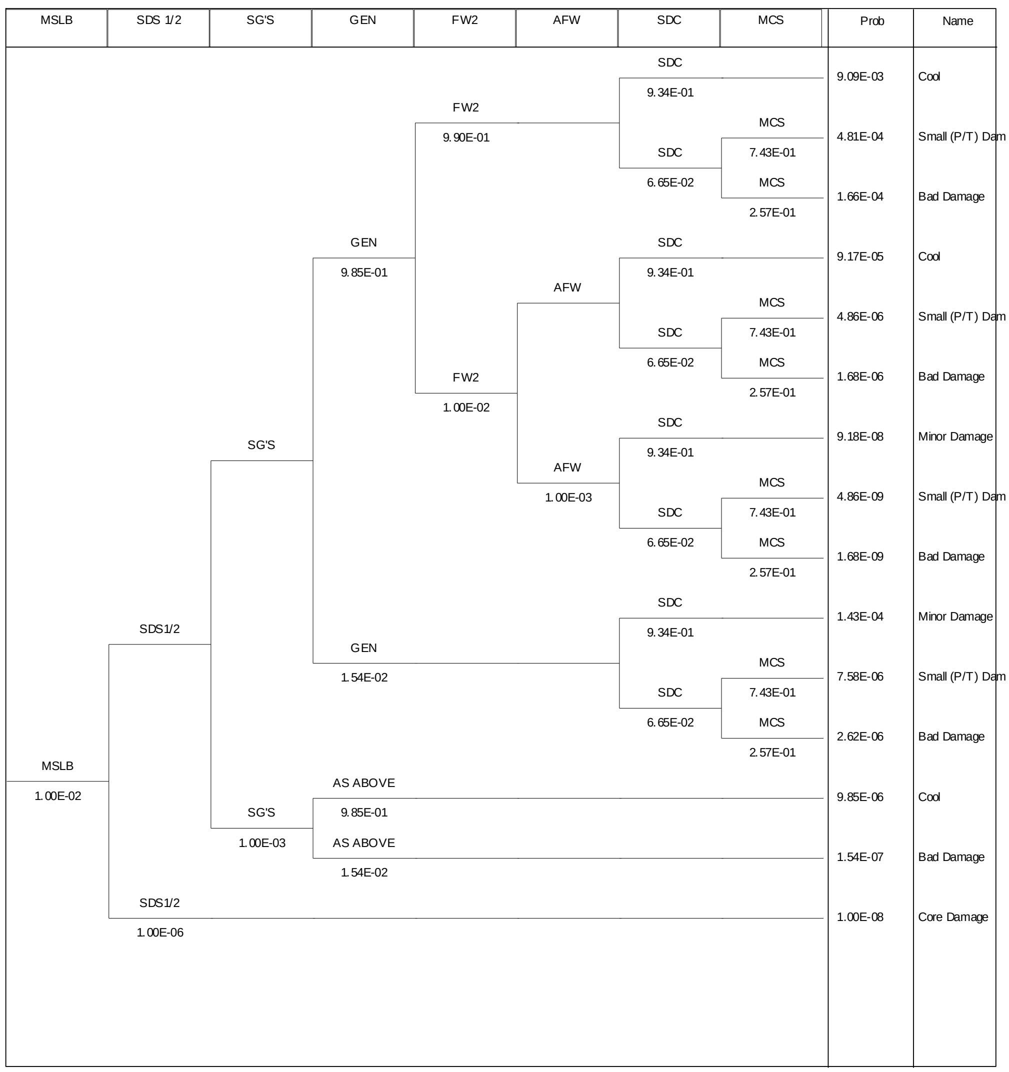
4.2 Main steamline break event

In a main steamline break accident, the priorities are to (a) shutdown the reactor, and (b) remove heat from the core. Once the reactor is shutdown, the next concern will be to remove the decay heat. In the short term, the reactor systems will respond by attempting to establish thermosyphoning between the steam generator and the core. Thermosyphoning is a process whereby decay heat is transferred by thermal convection to the steam generator: by the differences in density, the steam in the core rises to the cooler steam generator, while the condensed and heavier coolant falls to the core. The accident sequence and mitigation processes in both reactors will be presented here.

4.2.1 MAIN STEAM LINE BREAK – CANDU

Table 5 is an event tree for a Main Steam Line Break (MSLB) in CANDU. Initiating event: MSLB occurs in turbine room.

Table 5: Event Tree for Main Steam Line Break CANDU



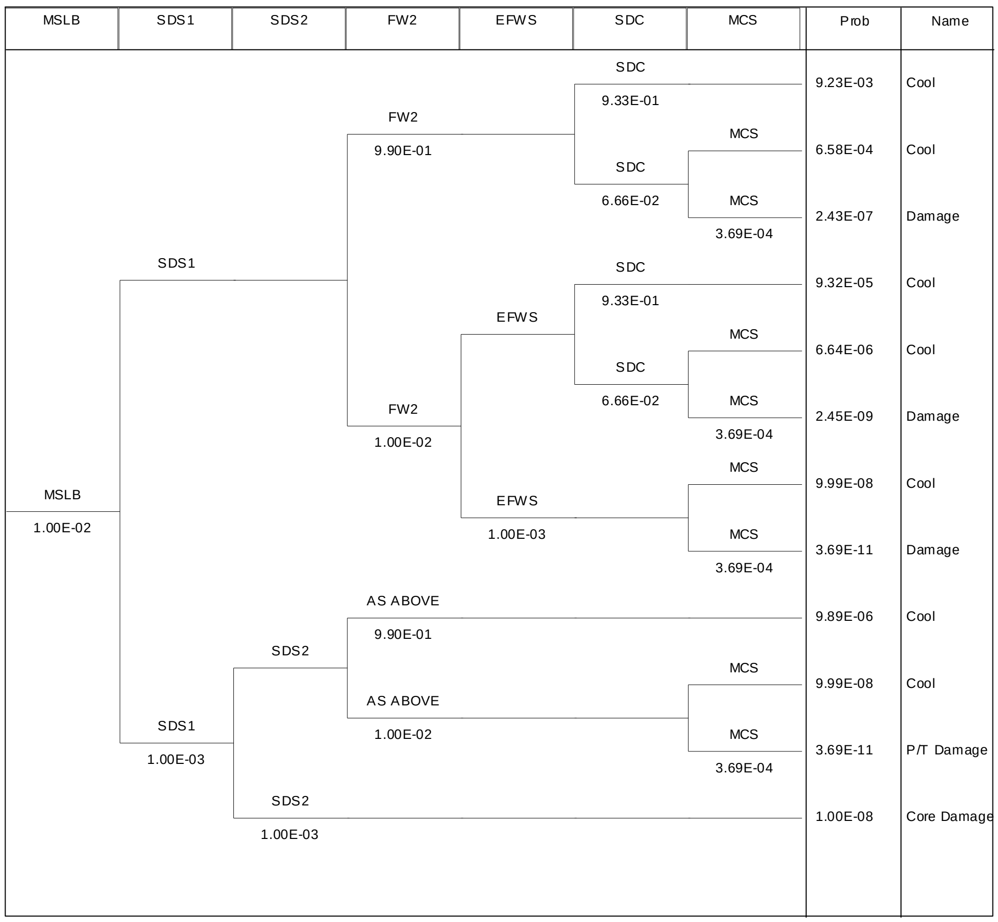
4.2.2 MAIN STEAM LINE BREAK – SCWR

Table 6 is the progression of a MSLB in SCWR. Initiating event: Main Steam Line Break occurs within containment. This analysis assumes that there is the IRWST; the other source of water available for emergencies is the Emergency feedwater system (EFWS)

1. The break means the turbine will trip to prevent it from overspeeding. *Event: Turbine Trip*
2. With the turbine trip, the reactor will shut down. *Event: Shutdown systems 1&2 (SDS1, SDS2)*

3. Again, the steam in containment may limit the number of systems that are able to function. If the Group 2 feedwater system is available, the coolant may be pumped to the core to keep it cool. *Event: Feedwater system 2 (FW2)*
4. A path will be established between the IRWST and the core for passive heat removal. The heat exchanger in the IRWST removes decay heat from the core by natural circulation, creating a form of thermosyphoning. Otherwise, the EFWS can be used as a water supply to cool the core. *Event: Emergency Feedwater system (EFWS)*
5. If that does not work, and also for long-term cooling, the SDC will be called upon to cool the core. *Event: Shutdown cooling (SDC)*
6. The moderator is still the backup decay heat removal system and can reliably remove heat from the core using the SCWR's passive moderator cooling system. *Event: Moderator cooling (MCS)*

Table 6: Event Tree for Main Steam line break in SCWR



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There are other undeveloped sequences for alternatives to supplying water to the steam generator to establish and maintain thermosyphoning for short-term cooling in the CANDU reactor. For example, the Boiler Emergency Cooling System (BECS), or the Fire water can be drawn on to replenish the steam generator supply. These alternatives are not considered in this analysis but acknowledged that there are other sources of water available. These sequences may be analyzed separately in future work. The Boiler Emergency Cooling system is lost in the SCWR.

The summed probability of damage to the CANDU from this accident is 7.09E-05/yr while the probability of damage to the SCWR using this event tree gives 2.16E-07/yr. It can be seen from the event tree that the worst reliabilities come from the SDC and the feedwater system FW2. The reliability of the SDC should be higher in the SCWR because fewer backup systems are available to mitigate the MSLB accident

For the SDC, the main contributors to poor reliability are the heat exchangers and the pumps. It was aimed to keep the reliability of all equipment ≥ 0.01/yr. The failure rate of the pumps was changed from 0.229/yr to 0.01/yr and that of the heat exchanger changed from 0.035/yr to 0.01/yr. This meant the SDC failure rate improved from 6.65E-02/yr to 1.29E-02/yr. That improvement made the probability of damage to the fuel or cladding of the SCWR from a main steam line break to be 3.38E-08/yr, thereby increasing its reliability.

5. Discussion

The aim of this work was to quantify the improvements in safety of the CANDU-SCWR by determining the improved level of risk and reliability displayed by the systems important to safety. Such analyses allow designers to focus more on systems that are most sensitive to failure and introduce more redundancy for the components or systems with low reliability. The study also sought to determine if the SCWR, as a GEN-IV reactor, had a reliability at least an order magnitude (or two) better than the generic CANDU systems.

The tool used for the analysis was the Probabilistic Risk Assessment, which is useful as it allows an analyst to conduct a sensitivity analysis on the system under consideration to know which components have more of an impact on the overall system's reliability. Comparisons were made to the response of the systems important to safety following two accidents in the reactors – a main steamline break and loss of forced circulation. The SCWR compared favorably with the CANDU in that it showed less risk of damage to the fuel and cladding. For the loss of forced circulation event, the SCWR showed at least an order of magnitude better reliability for damage to the fuel or cladding. The same occurred for the main steamline break accident. Also, when arbitrary adjustments were made to the systems that had a higher failure rate – in the form of ensuring all components had a failure rate ≤ 0.01/yr – the systems showed increased reliability. We believe such systems are good places to start to seek adjustments and improvements in the SCWR safety system to ensure that this reactor has displays better safety metrics than the previous generation reactors.

Current work includes attempts to analyze the impact of the loss of such systems and components as the steam generator and the BECS in other accidents, comparing the SCWR to the CANDU's safety and risk. Furthermore, from different fault trees constructed, it was noticed that the pumps have a fairly low reliability. It would be desirous to increase the reliability of this vital piece of equipment. Perhaps a different kind of pump is required, such as the canned motor pump – a centrifugal pump which may have a better reliability. This pump does not have a dynamic shaft seal, nor does it have couplings or ball bearings [9]. This pump can eliminate a seal LOCA arising from loss of seal cooling. The auxiliary fluid systems that support this type of pump are much simpler than the ones for a shaft seal-type pump.

The next step of this work will be to determine the consequences of accidents numerically, for instance the dosage received by the population from certain accidents. Having obtained frequency of occurrence from event trees, the Risk Analysis can be completed.

6. References

[1] P.G. Boczar et al., “Reactor physics studies for a pressure tube supercritical Water reactor (PT-SCWR)”, Proceedings of the 2nd Canada-China Joint Workshop on Supercritical Water-Cooled Reactors, Toronto, Ontario, Canada, 2010 April 25-28.

[2] W. Zheng et al., “Assessment of corrosion and stress corrosion cracking results for SCWR development: gaps and needs”, Proceedings of the 2nd Canada-China Joint Workshop on Supercritical Water-Cooled Reactors, Toronto, Ontario, Canada, 2010 April 25-28.

[3] Snell, V.G. and Garland, W, “Nuclear Reactor Safety Design”, Course notes, 2009.

[4] M. Modarres et al, “Reliability engineering and risk analysis”, CPC Press, 2010, pp. 15-20.

[5] U.S. Nuclear Regulatory Commission, “Probabilistic risk assessment procedures guide”, NUREG/CR-2300, 1982.

[6] U.S. Nuclear Regulatory Commission, “Reactor safety study – An assessment of accident risks in U.S. commercial nuclear power plants,” NUREG-75/014, 1975.

[7] IAEA, “Component reliability data for use in probabilistic safety assessment”, IAEA-TECDOC-478, pp. 96-289.

[8] A. Blanchard, “Savannah River site generic data base development”, WRSC-TR-93-262, Rev.1, 1998.

[9] DPPL site: http://www.cannedmotorpumpsindia.com/basic-canned-motor-pump.html, Accessed Oct. 22, 2010