RISK ASSESSMENTS AND REGULATORY CONCERNS FOR CANADA'S GEN-IV REACTORS

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Abstract

The Generation-IV reactor design Canada is working on is the Supercritical Water Reactor. It leverages the reliable CANDU fuel technology yet operates at temperatures and pressures that far exceed the traditional CANDU's operating conditions. The safety regulations of Canadian nuclear industries are examined in relation to the Supercritical Water Reactor to assess if the regulations require modifications to account for the new reactor type. Also necessary is a revised safety and risk assessment of this nuclear power plant. This paper highlights some of the considerations that will form the basis of a Probabilistic Safety Assessment of this new reactor.

1. Introduction

The Generation-IV (GEN-IV) reactors are slated to be built worldwide in the next few decades. Six reactor technologies have been selected for research and development under the international GEN-IV program. GEN-IV reactors are aimed at having superior safety by incorporating additional passive safety concepts. They are also expected to offer significant advances in operational performance, fuel cycle sustainability, and relative cost of energy. Canada is a lead nation in the design of one of the GEN-IV reactors – the Supercritical Water Reactor (SCWR). The SCWR design is based on the existing CANDU pressure tube configuration.

2. Canada's Generation-IV reactor design

Canada's SCWR can be considered as an advance of the CANDU-type reactor design, with the most important difference being the operating conditions, the fuel design, and the coolant type. The SCWR will still be a pressure tube type reactor using low-temperature heavy water (Deuterium oxide) as the moderator, but light water as coolant. As in the CANDU, online refuelling is possible but it is not required in the SCWR design [1]. It is expected that the next generation CANDU design will have a higher thermal efficiency and improve the economics of the reactor, primarily through capital cost and construction schedule reduction. Like the other GEN-IV designs, it is expected that the SCWR will use more passive safety systems than the traditional CANDU. An inherent safety feature of the SCWR is the passive moderator/shield tank heat sink surrounding the core.

Some of the comparisons between the SCWR and the CANDU Classic are in Table 1 below. One feature that is radically different from the traditional CANDU design is the use of light water as coolant rather than heavy water. Replacing light water as coolant, coupled with the reduced lattice pitch to improve certain neutronic characteristics, significantly reduces the cost of heavy water in the SCWR. This decrease in heavy water inventory should provide significant savings to the operating cost. Another radical difference between the traditional CANDU and the SCWR is that the SCWR will use enriched fuel. There will be benefits such as giving higher burnup, reduced quantity of spent fuel as well as lower fuel cycle costs from the fuel choice. The SCWR design Canada is considering will greatly increase the thermodynamic efficiency relative to the CANDU's. The proposed reactor will also be direct cycle. Since there will be no change of state in the coolant, the design eliminates the need for a pressurizer, steam generator and related secondary systems.

Feature	**SCWR**	**CANDU**
Rating (Mwe)	900-1140	740
Moderator	Heavy water	Heavy water
Coolant	Light water	Heavy water
Coolant pressure	25Mpa	10Mpa
Outlet temperature	625°C	310°C
Fuel	Slightly enriched uranium	Natural uranium
Lattice pitch	Smaller	Larger
Thermodynamic cycle	Direct	Indirect
Thermodynamic efficiency	40-45%	28-30%

Table 1: Comparing the CANDU to the SCWR

Water's supercritical temperature and pressure is 374°C and 22.1MPa. Since the SCWR coolant will be above these conditions (system pressure of 25MPa and outlet temperatures up to 625°C), the SCWR's design introduces unique avenues of risk. For instance, the jet impingement of a broken pipe will be significantly higher. Therefore, new risk and safety analyses need to be done for this new type of reactor. The SCWR is being based on supercritical water fossil fired plant experience, so the technology is not entirely new. It is expected to have enhanced passive safety systems, but even passive safety systems produce challenges of their own in terms of testing them to ensure they have the required performance reliability.

Passive safety systems are desirable because they are normally based on natural phenomena such as gravity and heat transfer methods such as convection and conduction. 'Passive system' means that electric power is not needed nor is mechanical forced power such as pumps. Therefore, passive systems are expected to have higher reliability and functionality. We anticipate that as a GEN-IV reactor, the SCWR will have passive safety features such as natural circulation for decay heat removal, and perhaps the system will have less valves and piping, thereby reducing the potential areas for failure. However, 'passive' does not mean that the system will be fail-proof. Passive safety systems can be difficult to test, and therefore it is hard to build up a reliability database. A lot of safety analysis will need to be done on the reactor to develop a good understanding of the thermalhydraulic system. If the SCWR is similar to the ACR (Canada's most recent CANDU design), it will have a reserve water tank. The reserve water system can be used under various accident scenarios for heat removal. For instance if there is an earthquake, the emergency water can be fed into the system to remove decay power heat. Also, under a loss of Class IV power, the decay heat can be removed using the auxiliary feedwater system. Being passive though, this safety system can still fail. For instance the reserve water tank can spring a leak. So several tests should be done to ensure the passive safety systems are very reliable.

3. Risk and safety assessments

Different methods can be used to analyse item failures, such as Statistical, Probabilistic, or Deterministic methods. Risk assessment is a method to identify, quantify, and evaluate failures or hazards. The quantitative aspect deals with estimating the probability and the consequences of the hazard occurring. Estimating the likelihood of a hazard's occurrence depends in large part on the reliability of the system's components, the system as a whole, and on human-system interactions [2]. The evaluation aspect of risk assessment can entail the risk-benefit or cost-benefit analysis.

Our research is conducting a Probabilistic Safety Assessment (PSA) as well as Probabilistic Risk Assessments (PRA) of the proposed SCWR. For both the PSA and PRA, the tool we will use for analysis is the Computer-Aided Fault Tree Analysis tool (CAFTA). The probabilistic risk assessment first aims to identify the types of accidents that could occur and their frequency of occurrence [3]. For each accident, an initiating event is defined as well as the subsequent events that could occur during the propagation of the initiating event. The PRA also quantifies the consequences of the accident.

PSAs and PRAs are being used in the nuclear industry currently. For instance probabilistic analysis methods were used to analyse the failure of the digital feedwater control system of a PWR [4]; PSA was done on a loss of regulation accident in a CANDU [5]; and the Analysis of Dynamic Accident Progression Trees software was used to improve the accounting of uncertainty between Level 2 and Level 3 PRA analysis [6]. Some safety concerns that are currently being researched in the nuclear industry include the possibility of a hard-to-extinguish Zirconium fire starting in the spent reactor fuel pool [8], and the susceptibility of the CANDU-SCWR to dynamic instability due to the sharp variation of fluid properties at water's supercritical point [1].

One way to evaluate the risk of an undesired event is to define risk R as the product of the frequency of the event F and the consequence of the event C:

$$R = FC \tag{1}$$

Risk assessment is necessary to determine the accident sequences that could lead to system failure and, if possible, remove the weakest links of the system [3]. That would allow changes to systems to be made in order to improve the plant's availability. Risk assessments are also valuable for pictorially viewing the risk involved in each stage, such as in a fuel cycle. A risk assessment can be done on the entire life cycle of the fuel– from the mining and milling operations to waste disposal – or the risk assessment could simply cover the nuclear plant operations. Our work will only look at the nuclear power plant (NPP) operations, but it would be interesting in the future to extend the safety and risk analysis to processes prior to the fuel arriving at the plant.

Fault trees and Event trees are pictorial and the analysis of these trees can be used to evaluate the expected frequency of an undesired event leading to certain consequences. WASH-1400 considered such consequences as early fatalities, latent cancer fatalities, and property damage. These are common datums used in the nuclear industry to estimate consequences of the undesired event under investigation.

In WASH-1400, the hazard to be avoided was the dispersal of radioactivity. Therefore, the radioactive releases from the core were identified for each accident sequence, and the consequences like deaths and property damage were estimated from the amount of radioactivity released.

An important issue in constructing an event tree is accounting for the timing of events. Sometimes, the failure logic will change depending on the time at which the events take place, for instance in the operation of the Emergency Core Cooling (ECC) system [3]. When the event tree has been made with enough information, the accident sequence must also be properly defined in order to calculate the consequence of each event. The Event tree in Figure 1 below is an example. It allows an analyst to determine the probability of Core Damage (CD) or Large Release (LR) given probabilities of failures of some of the safety systems, such as the CANDU's two shutdown systems (SDS1 and SDS2), the ECC, and the containment.

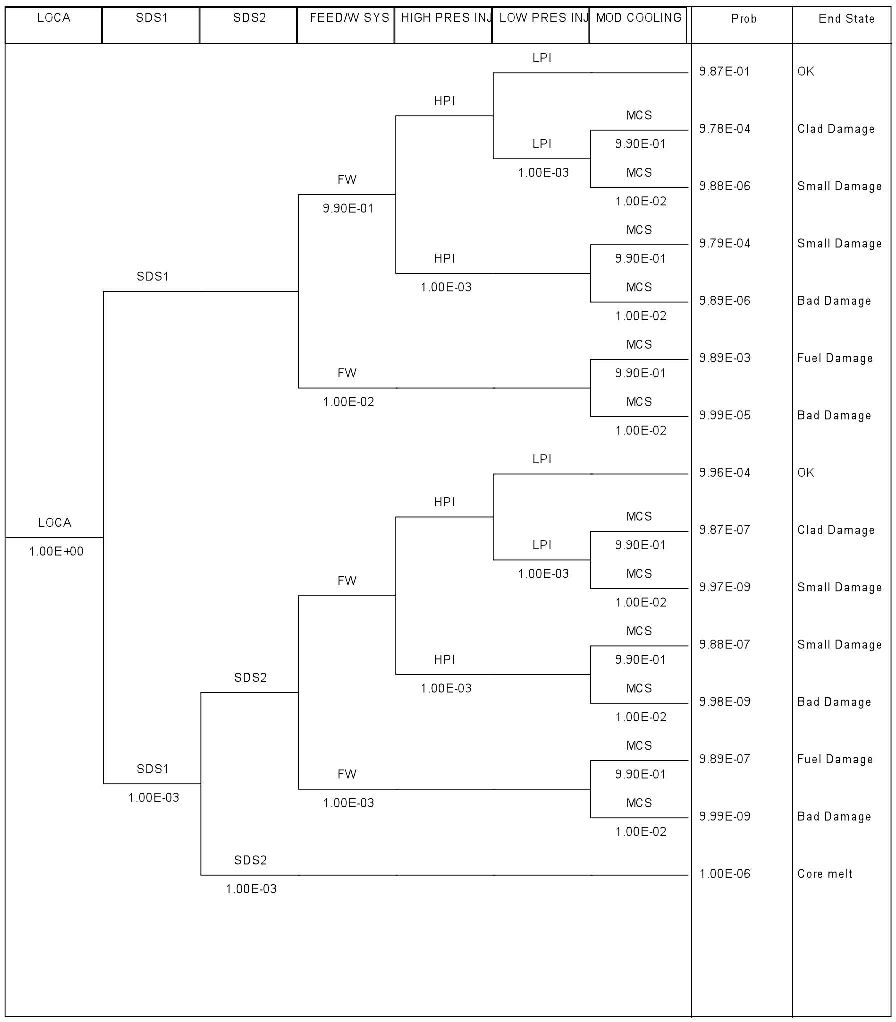

Figure 1 CAFTA-generated event tree of small LOCA

3.1 Safety assessment for SCWR

This study's PSA will measure the SCWR's safety by the four safety functions required in a nuclear reactor:

a) Shutdown the reactor
b) Remove decay heat
c) Contain any radioactivity
d) Monitor the state of the plant

A: Reactor Shutdown

CANDUs have two separate shutdown systems: Shutoff rods and poison injection. The Canadian nuclear industry regulator set the required unavailability of shutdown systems for the CANDU to be at most 1 failure per 1000 demands. The CANDU shutdown systems typically meet an unavailability rate of 10^{-4} however [7]. The two shutdown systems are geometrically as well as functionally separate. The Regulatory Document R-8 that outlines the requirements for a CANDU shutdown system mentions that each CANDU reactor must have "two independent and diverse shutdown systems", and the shutdown systems "as far as practicable, shall be of diverse designs and shall be physically and operationally independent from each other, from process systems and from other special safety systems" [8].

The CANDU shutdown systems have typically been designed to sense an accident using at least two diverse trip parameters. For instance the trip parameters for a small loss of coolant could be low Heat Transport System (HTS) flow, low moderator level, and low HTS pressure, or trip parameters for loss of forced circulation could be high neutron flux, low HTS flow, and high HTS pressure. But the new regulatory document RD-337 seems to give allowance for reactor trip without a second (or backup) trip parameter. Such an allowance is given in cases where a 'direct trip parameter' is available [11]. (A direct trip parameter is defined as a value based on direct measurement of a specific challenge to the derived acceptance criteria and, if applicable, a direct measure of the event [11].) Therefore, when a direct signal exists, reactor shutdown will be initiated for all AOO's and DBA's.

For reliability testing purposes, to avoid overstressing the systems, the logic is tested more often than the mechanism itself. For instance, if the demand availability of the shutdown system is more than 0.999 (as stipulated by R-8) – which effectively requires the system to always be available except for 8 hours per year – it would mean testing the system so frequently that the power plant would often be unavailable for operation. In addition, the plant must then be restarted quickly to avoid Xenon poison out. To perform less frequent tests therefore, one should test the logic that would trigger the shutdown system.

CANDU Classic has 3 logic channels, and a two-out-of-three vote is enough to trigger a safety system or initiate reactor trip. So a single channel can be tested without tripping the reactor. The ACR and modern Light Water Reactor designs use two-out-of-four voting logic for reactor trip. This implies that two out of the four instrumentation channels can trip the reactor. The two-out-of-four voting logic provides for one channel to be out of service (and left un-tripped) while maintaining a safe shutdown system. Thus, the operator avoids spurious trips since not only one more channel is needed to be fired to trip the reactor. However, there is an extra cost due to the extra trip channel – 33% more for instrumentation, maintenance, and testing [7].

Since the ACR is the most recent CANDU reactor offering, we expect the SCWR will also have two-out-of-four voting logic for reactor trip, allowing safe operation of the plant even if one channel has failed and is offline while being repaired.

B: Heat Removal

Under normal operating conditions the main steam and feed water system remove the heat for the CANDU reactor operating. The HTS transfers heat from the reactor core to the secondary coolant through the steam generators. Thus the steam generator is a heat sink for the reactor. Also, in a small LOCA, steam generators are heat sinks for primary coolant. In the SCWR, the steam cycle is a direct cycle, thus there is no use for a steam generator. If there is a sudden loss of heat removal from the secondary side of the CANDU Classic, the main steam safety valves will send the steam to the atmosphere. It was possible on a loss of heat removal from secondary side (e.g. a turbine trip) to dump steam directly to the condenser, bypassing the turbine.

Since the steam generators are no longer available as one of the heat sinks when assessing a Loss of Coolant Accident (LOCA) in the SCWR, we anticipate that the decay heat removal systems are going to have to operate at a higher performance level or require more reliability than before to remove decay heat as a backup to the system. The moderator and the shield tank will be a heat sink to prevent fuel failure. The moderator can remove heat from the HTS in severe accidents and the shield tank will delay the progression of core melt.

C: Containment:

The containment is a safety system which forms an envelope surrounding the systems that contain fission products and should be leak-tight in an accident scenario. Due to the high temperature and pressure of the SCWR, this requirement is even more vital given the potential for fission products to travel faster and further if an accident results in release of radioactivity. Containment is for shielding, a barrier for external events, and for hydrogen control in the reactor building. An important inclusion in RD-337 with regard to containment design is that nuclear power plants in Canada must consider malevolent acts when designing the containment, thereby mitigating severe accidents. So the containment structure must remain a continuous leak-tight envelope in all the plant operating conditions all potential hazard situations, from AOO's, to DBA's, severe accidents, and malevolent acts [11]. The containment design should also provide controlling systems in case of the release of fission products, hydrogen, oxygen, or other substances into the containment area.

D: Monitoring

As is the practice in monitoring CANDU's, we expect the SCWRs to be monitored primarily from the Main Control Room (MCR). During most accidents, safety functions such as shutdown, heat removal, and containment can be performed from the MCR. A Secondary Control Area (SCA) will be provided in case an incident such as fire, hostile takeover, or an airplane crash makes the MCR unavailable. If the ACR is used as a reference, we expect the SCWR's MCR and essential equipment will be qualified for post-seismic activities. The MCR in the CANDU Classic is seismically qualified, but certain equipment in the MCR is not. Hence, if an earthquake occurs on a CANDU site, the operators must evacuate to the SCA and perform their duties from there. The SCA will be equipped with the instrumentation and controls that are necessary to bring the plant to a safe shutdown state.

Nuclear instrumentation is provided to allow automatic control of reactor power and flux shape and to monitor localized core behaviour. Conventional instrumentation provides signals for the control and display of other plant variables as well. The instrumentation used in the safety systems incorporates triplicated signal channels and two-out-of-three logic which provides the required reliability for valid trip signals while giving the level of redundancy required to ensure that single component failures will not cause spurious operations. Moreover, the safety systems operate separately from the control systems.

The software used to control the shutdown system of the power plant must also be subject to safety measures. In CANDU, this is done by requiring strict independence between the software engineers who design each shutdown system [7]. This precaution might help in reducing software failures and also avoid possible common cause failures. (A common cause failure occurs when multiple components or systems fail due to a single secondary event.) Software has been used for operating shutdown systems in recent CANDUs; assuming the trend continues, software is going to be used for the SCWR systems. A stipulation from RD-310 is that the computer codes used in safety analysis be "developed, validated, and used in accordance with a quality assurance program" that meets the Canadian Standards Association N286.7-99 requirements.

3.2 Safety issues for SCWR

Canada's Generation-IV National Program has identified three areas that have the most pressing research need for SCWR. These are materials research, chemistry research, and thermalhydraulics and safety research [12]. Some of the design and operational areas that we identify as being safety concerns are:

a) Since the supercritical steam will go directly from the reactor to the turbines – no steam generator necessary in separate heat transfer loop – the system components that use steam, including the turbine, will have to be shielded, that is be inside the power plant Containment. This is because the steam will get irradiated after passing through the reactor core, thus passing harmful radiation to the turbine and other components.
b) Materials for the core need to be properly researched and tested. So far, the SCWR can draw on the techniques and processes of the fossil-fuel power plants (particularly the coal plants that use supercritical water). So some components such as pipes and turbine-generators for the nuclear plant will not be hard to obtain as they are already on the market. But the in-core elements in a CANDU have not been subjected to the temperature and pressure the SCWR will operate at, so design of fuel bundles, cladding, and pressure tubes may have to be redesigned. Or at least the material fabrication and testing such as thermal stress tests needs to be done for the fuel components. At present, the usual cladding material for CANDUs – a Zirconium alloy – is not going to be used in the SCWR because the material shows corrosion and creep at SCWR temperatures [1].
c) If a LOCA occurs in the small diameter piping for instance, the consequences will be more severe in the SCWR than in the CANDU because of the operating conditions of the SCWR – the coolant is flowing at about 25MPa through pipes as opposed to the CANDU Classic's 10MPa pressure. Therefore, the consequential damage from pipe whip or from jet impingement forces would be significantly higher in an SCWR.
d) Primarily, a nuclear power plant operator wants to prevent release of radioactivity. One part of the prevention process is to control water chemistry to minimize corrosion and transport of corrosion materials and radionuclides. Operations from fossil-fuel supercritical water plants indicate that there is a great risk of corrosion products being deposited on fuel

cladding surfaces [12]. These corrosion products may also become irradiated when they are being transported by the coolant, and if they are deposited on surfaces outside of the core, there is the potential of a release of radioactivity. The steam generator has typically been the repository for the material such as particulate that circulates with the coolant. The material originates from the corrosion on the primary side, and the material is usually deposited on the steam generator tubes. Balancing the water chemistry could reduce the feeder pipe corrosion; controlling the pH will minimize the dissolution of iron. To combat the dissolution of iron, the CANDU HTS pH specification has been set to 10.2–10.8 [8]. But other components might get corroded, especially with higher temperature and pressure.

e) Though not yet proven, we also expect that the probability of pipe failure will be increased with the operating conditions of the SCWR.
f) Another scenario we look at is a station blackout or loss of Class IV power. The consequence of this loss of power may be more severe in SCWR than CANDU Classic if not attended to quickly because the absence of a steam cycle means there is no secondary heat sink to cool the fuel while power is restored or pumps are brought back online.

4. Canadian nuclear industry regulations

The final aspect of the study is examining the regulations that will be pertinent to the SCWR. RD-337 is a recent regulatory document which sets the expectations for the design of new water-cooled power reactors in Canada. The regulatory document issues high-level design safety requirements, having revised the safety requirements from previous documents. Some changes include dose limits and accident classifications; these have been redefined. A single failure has been divided into two: The first is an Anticipated Operational Occurrence (AOO) which is a process outside of normal operations that is expected to occur at least once in the facility's lifetime but does not cause significant damage to items important to safety nor lead to accident conditions. The other accident a nuclear facility must be designed and built to withstand is the Design Basis Accident (DBA), an accident less likely accident to occur and which should not result in loss to the systems, structures, and components necessary to assure public health and safety. The dose limits for an AOO is 0.5mSv and for a DBA it is 20mSv [11].

4.1 Licensing and environmental assessments

There are three main steps set out by the Canadian Nuclear Safety Commission (CNSC) for a nuclear power plant licensee to perform an Environmental Assessment (EA):

- Preparing for the EA: This involves submitting an application for a license to CNSC, CNSC determining the type of EA required, and the Commission Tribunal deciding if the EA should proceed or be referred to a review panel
- Conducting the EA: This step entails doing the technical studies to evaluate the environmental effects of the project, developing methods to reduce or eliminate possible negative effects, and CNSC analyzing the technical study
- EA decision by the Commission Tribunal: CNSC prepares an EA report and consults the public on findings; the Commission Tribunal may hold public hearing based on environmental findings; if project is satisfactory to proceed, the applicant continues the license application process, and if approved, the CNSC issues the license

CNSC works with the Canadian Environmental Assessment Agency for this process. It is the Commission Tribunal that makes most EA decisions, however. An EA is valuable as it can predict the environmental effect of proposed initiatives before they are carried out. The EA process in Canada also gives opportunity for the public's participation in potential licensee's activities and for Aboriginal consultations.

An EA aims to identify the environmental effects of a project and proposes measures to mitigate any adverse effects of the proposed process. Thus, an EA can minimize or completely evade possible negative environmental effects from an activity, making environmental factors an important component of decision-making. For the nuclear industry, an EA is mandatory before a new build. Some of the benefits the industry gains by conducting an EA include creating more avenues for public participation, more opportunities to display government accountability, reduced risk of environmental disasters, and reduced project costs and delays [13]. In Canada, each province has its own EA legislation so the nuclear operating company would have to complete an EA in the province for which the new build is slated.

4.2 Some PRA and PSA regulations pertaining to SCWR

According to RD-337, the purpose of a PSA is to:

a. Identify accident scenarios with the potential for significant core degradation;
b. Demonstrate that a balanced design has been achieved such that no particular design feature or event makes a dominant contribution to the frequency of severe accidents, taking uncertainties into account;
c. Provide probability assessments for the occurrence of core damage states and major off-site releases;
d. Identify systems for which design improvements or modifications to operating procedures could reduce the probability of severe accidents or mitigate their consequences; and
e. Assess the adequacy of plant accident management and emergency procedures.

A PSA of a nuclear power plant is a comprehensive and integrated assessment of the safety of the plant. This assessment considers the probability, progression, and equipment failures in order to derive a numerical estimate of the safety of the nuclear reactor. A Level 1 PSA identifies and quantifies the sequence of events that could lead to loss of core structural integrity and massive fuel failures. The Licensee is required to perform a Level 2 PSA for the NPP. A Level 2 PSA builds on Level 1 PSA's results as a starting point, then goes on to analyse the behaviour of the containment, evaluates the radionuclides that are released from failed fuel, and quantifies the releases to the environment.

A PSA model must be, as close as possible, a reflection of the plant as it is built and operated. CNSC Standard S-294 requires that the PSA models be updated every 3 years or sooner if the plant undergoes major changes. The PSA we will conduct for the SCWR will include a sensitivity analysis and an uncertainty analysis, since all PSAs for NPPs in Canada must have a sensitivity analysis, uncertainty analysis and importance measures.

The PSA of a NPP can be done quantitatively by fault tree analysis. There are various methods and tools to perform fault tree analysis for nuclear plants. Some of these methods include PREP-KITT which was used in WASH-1400, FRANTIC, which like PREP-KITT can calculate the time-dependent instantaneous unavailability of a system, WAM, and CAT which uses ‘decision tables’ to describe possible output states as sets of inputs and internal operational or failed states [3]. The computer code we use for the PSA is called CAFTA. A requirement of the PSA as outlined in S-294 is that the methodology and computer codes to be used in a PSA for a Canadian NPP must first be approved by the CNSC [14].

5. Discussion

A constant criticism of the safety regulations of the Canadian nuclear industry is that they almost paralyse the Licensee by over-analysing the systems. There is a lot of conservatism (over-predicting consequences) built into the limits operating companies must adhere to. An advantage of these strict requirements is that the systems are designed with very high reliability. Thus, in AOOs or DBAs, the safety systems will most likely respond as needed to avert or mitigate disasters. However, it makes the operating company incur extremely high costs in developing tools and mechanisms that will function with extremely high reliability levels or according to guidelines for accident that will never materialize.

Should the regulations be relaxed? Should the dose limits from accidents or allowable occurrences of AOOs and DBAs be increased, thus reflecting ‘reality’ in NPPs? The current regulations and policies of the nuclear industry provide for safe operating conditions. Canada has benefitted from these strict rules as the safety history in the nuclear power industry can attest to – only 2 major accidents in almost 50 years of operation.

The newer regulations such as RD-310 and RD-337 were formalized in 2008 and override some of the older regulatory documents such as C-6 (under which the CANDU-9 station was licensed). RD-337 outlines the requirements for the design of new water-cooled power plants in Canada, whereas RD-310 sets the rules for safety analyses of NPPs. RD-337 is an enhancement of older regulatory documents and considers not only CANDU-type reactors but is open to any water-cooled reactor that will be built in Canada. That is important since the GEN-IV reactor Canada is working on will not operate exactly as a CANDU, having light water as coolant unlike CANDUs which are both heavy water-moderated and use heavy water coolant. The improvement of the RD-337 then is that it is technology-neutral.

Our main criticism of RD-337 is that it does not lay out its statements as firm requirements and obligations for the operating company to mandatorily follow. For most of the document, it does not use such words as ‘shall’ to outline the responsibilities of the operating company. RD-310 on the other hand proffers its report with definite terminology, e.g. “The analysis method shall involve the following elements…”, or “Analysis for AOOs and DBAs shall demonstrate that…” [16].

One of the purposes of this study was to determine if the regulations currently surrounding the Canadian nuclear industry are sufficient for the next generation of reactors expected to be built in the next few decades. The SCWR Canada is working on differs from the CANDU Classic, some of the differences highlighted in Table 1. Our analysis showed that the RD-337 caters to the current SCWR design – not having a steam generator for instance. In Section 8.3.1, the regulatory document mentions requirements and then adds the clause “... where applicable, steam generators, ...” giving allowance for reactor designs that do not include steam generators, such as the SCWR. Other regulations that we have investigated have guidelines that can apply to the SCWR. Some of the regulatory documents we studied are R-7, the Requirements for Containment Systems; R-8, the Requirements for Shutdown Systems; and R-10, The Use of two Shutdown Systems in Reactors. (It is not known now if the SCWR will have two shutdown systems like the CANDU does, but it is very likely to, seeing the success it has provided for Canada's reactors.)

It is not surprising that RD-337 makes room for the SCWR design: the Generation-IV International Forum was chartered in 2001 and plans for the SCWR began shortly after, while RD-337 was published in 2008. Because it is an entirely new reactor with different operating conditions – such as operating above the properties of supercritical water, 22.1Mpa pressure and 374°C temperature – an entirely new safety analysis needs to be done on the reactor.

Early work on our PSA shows that areas of interest will include decay heat removal following a loss of coolant accident, release of radioactivity from corrosion products deposited on out-of-core materials, and the consequences of pipe whip or jet impingement forces following a pipe break. Further PSA studies are ongoing to complete a full analysis of the risks that could be encountered with the Supercritical Water Reactor and the safety measures that need to be established to avert or mitigate such risks.

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