ASSESSING THE APPLICABILITY OF CANADIAN REGULATIONS TO THE SUPERCRITICAL WATER REACTOR

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Abstract

This paper is a review of some of the regulatory documents that bind the operation of nuclear power plants in Canada. The Supercritical Water Reactor is a new reactor that Canada is interested in building in the coming years. It is imperative that the regulations under which the reactor is operated are relevant to its operating conditions. This study examines some of the regulations the nuclear power industry adheres to, to determine their applicability to this new reactor under all plants states –normal operating conditions, anticipated operational occurrences, design- and beyond design- basis accidents scenarios,

1. Introduction

Canada has committed to the Supercritical Water Reactor (SCWR) as its Generation-IV (GEN-IV)reactor design because, among other features, it will have a much greater thermal efficiency than the CANDU currently used in the country. The SCWR being designed in Canada will be a CANDU-type reactor with heavy-water moderator but will differ from existing CANDU in that it will use light water as coolant, use either slightly enriched uranium or thorium fuel, and have a direct thermodynamic cycle. Of particular importance is the difference in operating conditions: the SCWR will have outlet temperatures up to 625°C and system pressure of 25MPa, compared to the traditional CANDU's 310°C outlet temperature and 10MPa pressure. These changes will result in thermal efficiencies on the order of 45 to 50%.

It has been recognized that since this reactor design is first of its kind, the current Standards, Policies, and Regulations issued by the regulatory authorities would not have accounted for unique features of the SCWR. Therefore, this study was embarked upon to identify the regulations that would be pertinent to the licensing of a new reactor and then to examine if the regulations – in their current form – would apply to the SCWR. If any were found not to make provision for this new reactor, recommendations would be made for modifications or enhancements to the regulations. This is in an effort to ensure the reactor is capable of being licensed in Canada to provide the much-needed benefits of higher efficiency and better sustainability, while operating safely and reliably.

The paper is structured into subsections which provide a summary of each of the major regulatory documents and framework and discusses the applicability of each to the SCWR design concept.

2. Background and Purpose

The key principle of the Canadian Nuclear Safety Commission (CNSC) with respect to guidelines for nuclear safety is that the nuclear power plants are to be designed and operated in such a manner that will protect the individuals, society, and the environment from harm. This general safety objective is categorized into two safety objectives: Radiation protection and Technical safety. Technical safety entails considering all possible accidents considered in the design; this includes those accidents that are of very low probability. The objectives of the Technical safety are to give every reasonably practicable measure to prevent accidents at the power plant, and to mitigate the consequences of these accidents should they occur.

The primary duty of a nuclear power plant Owner/Operator is to ensure that the plant is operating safely at all times. Tremendous effort is made to ensure there is minimal release of harmful radioactive fission products to the environment. In a reactor, there are five main barriers to radioactivity release from the fuel; sequentially, they are: the fuel pellet, the fuel sheath, the heat transport system, the containment, and finally the exclusion zone. The nuclear industry regulator is responsible for ensuring that all nuclear-related activities are carried out safely and that the staff and citizens in the vicinity of a nuclear power plant (NPP) are not negatively affected. One of the ways CNSC (the nuclear regulator in Canada) enforces these safe conditions is by establishing guidelines for operation in the form of Standards, Guides, and Regulations. A NPP is built specifically to provide electric power and so will have different rules applied to it than say a low-power facility such as a research reactor.

The regulations set out by the CNSC are applied to the nuclear power reactors that are in Canada. But as new types are built or brought into the country, these regulations may have to change. With the CANDU-SCWR, new regulations might be needed. That is what this study examines.

Some of the differences between the generic CANDU and the CANDU-SCWR were listed in the previous section. Other design changes that have been proposed are to have a direct cycle, thus eliminating the need for a steam generator, and to have the turbine within containment. Internal events will then be more of a concern because previously, the turbine was separated from the containment. (This design concept of the turbine being within containment might not be conflicting with the GEN-IV goal of better economics. It may not create a significant cost disadvantage of making containment larger than CANDU's if the turbine used is similar to a gas turbine, hence smaller than steam turbines.) With the increased modes for containment to potentially be compromised, there might need to be new rules by which the plant is built and operated, or changes made to the legislated maximum frequency of incidents that would affect the containment. This is especially important because breaching containment is one of the last lines of defence from radioactive elements harming the population. A turbine failure due to a blade breaking off during operation could breach Heat Transport System and containment. In the traditional CANDU, that failure could not have been a severe accident, but for the SCWR, it could be a severe accident scenario when the turbine is within containment as the current pre-conceptual design calls for. The extent of this event is now such that it could breach 4 of the 5 barriers the NPP has to the release of radioactivity.

Another factor to be considered is the dose effect of turbine failure/rupture within containment. This study is being done in conjunction with another [1] which is looking at the reliability of components such as the turbine, and to assess if an accident with the turbine would reduce the effectiveness of the containment as a barrier to radioactive fission release. That study is also examining the safety criteria necessary to ensure it conforms with the set standard for Anticipated Operational Occurrences (AOO's) in terms of frequency of occurrence and dose release. (From Regulation RD-337, the whole-body dose limit for AOO's is 0.5mSv, and for Design Basis Accidents (DBA's), it is 20mSv [2].) Furthermore, the energy of a failed pressure tube could fail others in its vicinity by the force of the explosion. An extension of this study will be to determine if this can happen at supercritical conditions in the core, since Regulation R-8 states that a failure should never propagate to create others [3].

Therefore, the aim of this paper is to determine if the requirements for safety and licensing are met, and to identify where there are gaps in the regulations, and perhaps difficulties, in confirming the regulations are relevant to the new reactor. For instance the SBLOCA is categorized as an AOO in the CANDU with a 10^{-2}/yr limit on frequency. But in the SCWR, perhaps it will not be thus categorized. Also, with the presence of the passive moderator system that should keep the core cool under various accident scenarios, a beyond design basis accident (BDBA) like a LOCA + LOECI may now be classed as a DBA. So the limits on the occurrences of incidents would need to be reassessed and restated.

The regulations are to ensure the public is kept safe from ionizing radiation generated during the plant's operations or other power plant accidents.

3. Regulations and standards review

This section examines the Canadian regulations pertinent to licensing a new reactor to determine if they would be applicable to the SCWR.

3.1 Regulations

These are some of the regulations that might be pertinent to the licensing of a new NPP in Canada.

3.1.1 RD-346: Site evaluation for new nuclear power plants

This regulation provides general criteria for site evaluation. The regulation states that the main objective for site evaluation is that the NPP will not "create unreasonable risk to the public or to the environment" [4]. The licensee is responsible for identifying and prioritizing the risks associated with the site's characteristics and external events.

In this Regulatory document, some of the criteria for evaluating an NPP site are:

a) Evaluation against safety goals
b) Consideration of evolving natural and human-induced factors
c) Evaluation of the hazards associated with external events
d) Determination of the potential effects of the NPP on the environment
e) Consideration of projected population growth in the vicinity of the site, and emergency planning that takes those projections into account.

The regulation explains how the proposed NPP site is evaluated against hazards e.g. natural, human-induced, and external events. A good way to analyse the NPP site is to consider these events occurring as well as the direct effect of an event that would affect the safety of the plant such as an earthquake resulting in a main steam line break.

Sometimes external- or human-induced events may combine with natural events or hazards; the regulation gives examples of this occurrence, like the simultaneous aircraft crash and a heavy snowstorm, or simultaneous oil spill and a fire. In selecting the accidents to be considered it is recommended that the list of existing accident sequences in operating CANDUs be reviewed and included in the SCWR bases, and also supplemented for any unique features of the SCWR design.

The list of accidents selected therefore includes: Loss of flow, fuel handling accidents, LOCA, electrical failure, reactivity-initiated accidents (RIA), secondary side breaks, and support system failures.

The NPP site is also chosen by balancing the needs associated with the plant – construction, operation, security – with social needs such as commercial and recreational use of the surrounding area. The licensee is expected to conduct risk modeling when developing the *Environmental Impact Statement* during the Environmental Assessment. This risk model can be re-evaluated over the lifetime of the plant and it is important that the initial design be risk-informed, so that downstream issues are minimized

The regulation includes plans for life extension of the plant through replacement and refurbishment and power uprate; so each new NPP has room to increase its output capacity. New regulations might also affect them so the site might have extra land or space for plant modification design. Since it is likely that a CANDU-SCWR would require a retube or some form of refurbishment during a 60 year life, some of the potential effects of extending the plant life are:

i) Additional conventional and radiological waste generated, which will in turn affect the handling, transport, and storage of waste
ii) Security and emergency planning might have to be revised because when there are more buildings, more threats are added as Design Basis Threats.

Finally, stakeholder consideration is required for the siting of a new NPP. Part of the stipulation of consulting with all the stakeholders is to establish the objectives and expectations of the consultation process. At this stage of the design process no action is required here.

Overall, RD-346 is a comprehensive and relevant regulation for the planned SCWR. It is entirely applicable to the SCWR, and no areas need to be changed to accommodate this new reactor.

3.1.2 RD-310: Safety analysis for nuclear power plants

The CNSC expects this regulation to be applied in new-build submissions. This regulation is for the deterministic safety assessment of an NPP. (The probabilistic safety assessment Standard is in S-294.) Safety Analysis is defined in RD-310 as "an analytical study used to demonstrate how safety requirements are met for a broad range of operating conditions and various initiating events" [5].

Events to be analysed are classified into three classes of events based on probabilistic studies and engineering judgement. The classes of events are:

a) AOO's, which include all events with frequencies of occurrence equal to or greater than 10^{-2} per reactor year
b) DBA's, which include events with frequencies of occurrence equal to or greater than 10^{-5} per reactor year but less than 10^{-2} per reactor year
c) BDBA's which include events with frequencies of occurrence less than 10^{-5} per reactor year

This study seeks to generate an interim list of events under normal operating conditions, AOO's, and DBA's. An AOO that can occur in both the CANDU and SCWR is a Loss of offsite power. As mentioned earlier, DBA's might change between those in the CANDU and those classed as such in the SCWR. DBA's that can be seen in the present CANDU and the SCWR are listed in Table 1 below.

Table 1: Selected list of accidents

CANDU	SCWR	Comments
DBA	DBA	–
Loss of coolant	Loss of coolant	–
Loss of regulation	Loss of regulation	–
Steam generator tube rupture	Steam main break	The SCWR will not have a steam generator
Loss of moderator cooling	Loss of moderator cooling	–
End fitting failure	End fitting failure	–

Apart from the quantitative references for the events, the operator needs to establish the qualitative acceptance criteria for the event. That way, the analyst will know (following the safety analysis) if the plant systems meet the target set. The ultimate goal is that the integrity of physical barriers is maintained and release of radioactive material is prevented following an AOO or DBA. The regulation suggests that quantitative (derived) acceptance criteria should be used for AOOs and DBAs to demonstrate that the qualitative acceptance criteria are met. These numerical targets are obtained from other experimental data prior to performing the safety analysis.

The results of the safety analysis are to meet "appropriate derived acceptance criteria" [5]. Given the large uncertainties in SCWR conditions and behaviour (for example heat transfer phenomena), it is critical that all uncertainties be quantified to the extent possible, and that design margins be included to account for future discoveries and experiments.

Every plant must have Safety Analysis Documentation. This Safety Analysis Documentation will be helpful for people who are not necessarily familiar with the plant yet need safety-related information on it. This record-keeping also helps another analyst to repeat the safety analysis done on this plant and thereby generate similar results. Furthermore, the Analysis Documentation will enable the analysis to be updated when new results become available. The new results are generated when the safety analysis is updated with posterior probabilities – knowledge after the reactor has been in operation for a while. Thus, the updated data may be site-specific; at least it would come from supercritical water reactors, and not from extrapolated CANDU values or values chosen based on expert engineering experience. Then the Safety Analysis Documentation is a better reflection of the plant in question.

A licensee has to review the safety analysis to ensure they meet the set objectives. The safety analysis is to be periodically reviewed and updated to account for changes in NPP configuration, conditions (including aging), operating procedures, research findings, and advances in knowledge and understanding of physical phenomena. Finally, if a different hazard is realized long after the initial analysis is done, these differences should be incorporated into the next ‘periodic update’ to keep the Safety Analysis Documentation current.

Overall, this regulation is relevant to the SCWR since the plant must undergo Safety Analysis, and the design method will incorporate results from this form of safety analysis (deterministic safety analysis).

3.1.3 RD-337: Design of new nuclear power plants

This regulation specifies the CNSC expectations for new water-cooled NPP designs. The CNSC drew on the principles in IAEA document NS-R-1 *Safety of Nuclear Plants: Design* in writing this regulation, and adapted those principles to align with Canadian practices. Since the regulation is intended to be technology-neutral, it is anticipated that most of RD-337 will be applicable to the SCWR concept.

A key element of a new design is to define the list of accidents to which the design must be able to mitigate the consequences. Table 1 lists some of the accidents that the SCWR might be exposed to and expected to withstand. A safety analysis will be done for those accidents to estimate, among other things, what systems will be necessary to stop or minimize the effects of the accidents. Then it will be determined if the components or systems respond with a reliability up to a stated goal. The RD-337 regulation gives a numeric reliability guideline for the design of a new reactor: “The safety systems and their support systems are designed to ensure that the probability of a safety system failure on demand from all causes is lower than 10^{-3}” [2]. Ref. [1] describes how such analysis for reliability is conducted on certain systems important to safety. Safety analyses are carried out to examine the plant performance for the four categories of plant states: a) Normal operation, b) AOOs, c) DBAs, and d) BDBAs, including event sequences that may lead to a severe accident. Most importantly, the results of these safety analyses are fed back into the design. This process – Risk-informed design – should prove invaluable to the Owner/Operator since the constant feedback loop can make the design process better and more robust. For instance, when the safety analysts discover an accident sequence that has a high probability of occurrence and whose mitigating systems have a low reliability, the designers can either insert a redundant system to cope with the accident or seek better materials to increase the reliability of the mitigating systems.

Various guidelines regarding the NPP design are given in this regulation; for instance it stipulates that the design management must ensure:

(i) Systems, Structures, and Components important to safety meet their respective design requirements. The SCWR will have specific design requirements, especially regarding the components’ safety and reliability. For example, one of the conceptual requirements is that the systems use more passive safety techniques for operation.

(ii) Safety design information necessary for safe operation and maintenance of the plant and any subsequent plant modifications is preserved. This means the design should identify the operating and maintenance requirements. Therefore each system should be reviewed for operability and maintainability.

(iii) Due consideration is given to the prevention of accidents and mitigation of their consequences
(iv) The results of deterministic and probabilistic safety assessments are taken into account
(v) The plant design facilitates maintenance throughout the life of the plant

The regulation also states that an independent peer review of the safety assessment would be performed before the design is submitted. The basis of the safety assessment, as the regulation informs, is the data derived from the safety analysis, operational experience, research, and proven engineering practices.

SCWR should definitely be built around the stipulations in this regulation. It provides beneficial guidelines for the operation and reliability of the safety support systems, and following these rules in design could contribute to the SCWR being a safer and more reliable reactor.

3.1.4 R10: The use of two shutdown systems in reactors

This regulation mandates that each reactor shall incorporate two independent protective Shutdown Systems (SDSs), unless CNSC approves otherwise. The issue of two SDSs was introduced to counter the effect of positive coolant void reactivity in CANDU. The positive coolant void reactivity means that in a LOCA, the reactivity of the core would increase thereby increasing power in the reactor. The immediate response necessary would be to shutdown the reactor to prevent damage to the core or structures. The second SDS was needed to ensure the CANDU reactor would have a high reliability of shutting down within a few seconds in accident scenarios. However, the SCWR in its current pre-conceptual design stage might actually have a negative coolant void reactivity. In that case, the requirement for two SDSs to compensate each other is muted because power will decrease on void of coolant. In case the Canadian design goes with both SDSs, the remainder of the regulation will be analysed as such.

Some design requirements of the SDSs mandated by this regulation include:

- Each of the two shutdown systems, acting alone to shut down the reactor, must also be capable of preventing failure of the primary heat transport system due to overpressure, excessive fuel temperatures or fuel break-up.
- Each of the two protective shutdown systems, acting alone, shall be capable of maintaining the reactor in a suitable subcritical shutdown state indefinitely or, alternatively for a period long enough to permit the protective shutdown system to be supplemented reliably
- Each protective SDS shall incorporate sufficient redundancy to ensure that no single failure results in the loss of its protective action.

This regulation might not be relevant to the SCWR since the SCWR might only require one fast-acting SDS if the coolant void reactivity is negative. But if the design uses two SDS's, all the stipulations of the regulation will apply to the new reactor.

3.1.5 R8: Requirements for shutdown systems for CANDU nuclear power plants

This regulation is also written for the CANDU NPP and so refers to two SDSs. Some of the availability requirements are:

a) Each SDS shall be designed such that the fraction of time for which it is not available can be demonstrated to be less than 10^{-3} years per year.
b) The design shall have sufficient redundancy such that no failure of any single component of a shutdown system can result in impairment of that system to an extent that the system will not meet its minimum allowable performance standards under accident conditions.
c) The design shall be such that it is not readily possible for an operator to prevent actuation of a SDS when such actuation is required.
d) As far as practicable, the SDSs shall be of diverse designs and shall be physically and operationally independent from each other, from process systems and from other special safety systems.

These requirements imply that if two SDSs are used for the SCWR, the combined unavailability will have to be less than 10^{-6} years per year. The third requirement above is also in line with the philosophy of the SCWR – being built with passive safety concepts. If the SDSs are designed such that an operator cannot stop them from operating once activated, the design will be drawing on the passive safety principles.

The SCWR is a CANDU-type reactor and so the directives of this regulation could be incorporated into the SCWR's design without modification.

3.2 Standards and guides

Other regulatory documents that are used for licensing a new NPP are standards and guides.

3.2.1 S-294: Probabilistic safety assessment for nuclear power plants

Items of note are the specific Probabilistic Safety Assessment (PSA) requirements. They include

- The NPP must perform a facility-specific Level 2 PSA
- PSA models should be as good a representation as possible the plant as built and operated
- PSA models must be developed using assumptions and data that are realistic and practical
- The CNSC must accept the methodology and computer codes used for the PSA
- The **PSA models must be updated every three years**, or sooner if the facility undergoes major changes

This Standard is relevant to the SCWR as part of its safety analysis will be done using PSA.

3.2.2 S-98 Rev.1: Reliability programs for nuclear power plants

S-98 is a document that describes CNSC requirements for systems and components reliability, and for the programs which are implemented at operating stations to track these issues.

The Standard states that the NPP needs to operate within certain stated bounds of overall safety [6]. This "overall safety" can only be maintained when the systems important to safety are: a) adequately capable of performing their purposes, and b) available to perform those purposes. For instance there will be clear goals such as the minimum reliability of continuously-running equipment and the availability of equipment that are called upon for specific situations (such as diesel generators).

As part of the drive to improve safety in the GEN IV designs, the systems important to safety in the SCWR will need to show an increased level of reliability. This regulatory document urges that a Reliability Program be established. This standard would be useful during all the phases of the reactor life cycle. Therefore it should be incorporated into the design and operating of the SCWR. In particular in the design phase, as part of the reliability program, it would be most important to:

a) Identify the components important to safety. Based on the existing design information from AECL these systems include:
    - Main heat transport system.
    - Special safety systems
    - Moderator system
    - Low power cooling system
b) Specify the minimum performance levels that must be attained by the systems important to safety
c) Identify systems which can be targeted for simplification, removal, and or replacement by passive systems.
d) Identify the possible common cause failure routes between systems.
e) Make provisions to prove that the reliability program is being implemented effectively

Such a reliability Program should be established in the SCWR operations, especially as there is not a lot of data on the response of this plant at the conditions it will operate at.

3.2.3 <u>G-144: Trip parameter acceptance criteria for the safety analysis of CANDU nuclear power plants</u>

G-144 is a regulatory guide; as such, it gives a recommended approach for meeting particular aspects of CNSC's requirements and expectations for licensing. This Guide focuses most of its attention on fuel sheath dryout and post-dryout effectiveness in existing CANDU's. This is because dryout can be taken as an acceptable alternative to fuel failure, and the resulting pressure tube failure, for safety assessments and monitoring. This Guide is written for CANDU NPP's, and though the SCWR is to be a CANDU-type reactor, some of its characteristics differ greatly from the traditional CANDU. For instance the SCWR will operate with supercritical water as coolant which does not experience dryout and as such fuel sheath dryout need not be a trip parameter in this new reactor.

One method of developing alternate trip parameter criteria for the SCWR is to first of all identify the mechanisms for fuel failure. Some of the fuel failure mechanisms are melting, fuel fragmentation, strain, and oxidation. Therefore, some alternate trip criteria which should be considered are:

a) **Strain level**: OECD's *Fuel Safety Criteria Technical Review* [7] states that the typical "conservative design limits" for fuel stress is around 1% yield or tensile strength at operating temperature; the limits for strain is 1% of the maximum circumferential elastic and plastic strain, and a maximum of 2.5% permanent axial and tangential strain at end of fuel life. However, the margins from these limits to actual failure stresses and strains should be obtained from the vendor's database, given the fuel, cladding, and burnup range.

b) **Oxidation and Hydriding**: Oxidation degrades material properties; as related to the reactor core, oxidation degrades thermal conductivity of the sheath. Therefore, the sheath loses its effectiveness to transfer heat. Hydriding, on the other hand, leads to embrittlement. These two effects become more important at higher fluence since the dependence on burnup is not linear. From reactor operating experience, the oxide thickness and hydride concentration values at end of fuel life are 100 micron and 500-600ppm respectively.
The 100 micron oxide thickness has been found to be the level at which there is a steep increase in the possibility of oxide spalling, which in turn gives room for hydriding and thus more oxidation. So perhaps these limits of 100 micron and 500-600 ppm could be used as limits for the sheath material in the SCWR. However, the SCWR is not expected to use the same Zirconium-2.5% niobium alloy as the CANDU since the temperature excursion expected for abnormal conditions yields temperatures beyond the operating envelope for Zirconium alloys. Hence, the CANDU-SCWR will use stainless steel cladding. This choice of sheath implies the oxidizing and hydriding limits might have to be revised.

In lieu of the complexity of determining sheath strain during all possible DBAs, it is likely that a more practicable limit be proposed for the design which ensures a large margin to fuel and pressure tube failures. It is proposed that such a criteria be based on a given fuel power ramp rate limit, a maximum sheath temperature, and for full power operation. Under start-up and low power operation it is likely that additional criteria will be required which can accommodate the sliding start-up-like procedures, low power cooling requirements, and maintenance. For the sheaths being considered the likely criteria will be 800°C for normal operation and 1200°C for accidents.

3.2.4 <u>G-149: Computer programs used in design and safety analyses of nuclear power plants and research reactors</u>

It is imperative that the analysis conducted to support the design, or licensing of a new design, be conducted using tools and methods that are qualified for the purpose. For example the mathematical equations used in a computer program must sufficiently reflect the phenomena and processes of the physical system they model.
G-149 provides guidelines for computer programs used in designing NPPs and used for analysing operational transients, incidents, or accidents. The guidelines cover such phases as the development phase of the computer program, the design phase, the code verification phase, the program integration phase, and the computer program performance validation stage.
G-149 is applicable to the design and safety analysis of the SCWR as following the guidelines provided herein could lead to robust programs being used for the development of this new reactor.

4. Discussion

This study is examining the regulations that will be pertinent to the SCWR. The SCWR's operating temperature and pressure present unique challenges for safety and licensing in Canada. Furthermore, CNSC needs to verify that these new power reactors are subject to safety standards and regulations that are appropriate, given the vastly different operating conditions.

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. 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 Darlington 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. The regulation being amenable to every water-cooled reactor is important since the GEN-IV reactor Canada is working on will operate somewhat differently than 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.

Some of the regulatory documents 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.) Most of the regulations – in their current form – were found to be acceptable to cover the SCWR for licensing. Table 2 below summarises the applicability of the regulations reviewed to the SCWR, identifying if any modification is necessary this new reactor.

But we wish to stress that as of now, the design requirements are not yet set and so the current and future work is to comprehensively examine the regulations to assess their applicability to the SCWR by:

- Generating a list of credible accidents
- Listing the systems important to safety (there may be more in the SCWR than in the generic CANDU)
- Establishing the design requirements
- Determining the limits for dosage and accidents (from sources such as RD-337)

Table 2: Applicability of Regulations to Supercritical Water Reactor

Ref.	Regulation	Application
R-8	Requirements for Shutdown Systems for CANDU Nuclear Power Plants	Yes
R-7	Requirements for Containment Systems for CANDU Nuclear Power Plants	Yes
R-10	The Use of two Shutdown Systems in Reactors	Maybe
P-223	Protection of the Environment	Yes
RD-337	Design of new Nuclear Power Plants	Yes
G-144	Trip parameter acceptance criteria for the safety analysis of CANDU Nuclear Power Plants	No
G-149	Computer programs used in design and safety analyses of nuclear power plants and research	Yes
RD-346	Site Evaluation for new Nuclear Power Plants	Yes
RD-310	Safety Analysis for Nuclear Power Plants	Yes

Acknowledgments

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5. References

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