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Application of the Most Recent Technical Standards to the Safety Analysis of Wolsong Unit 1 Nuclear Power Plant

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1. Introduction

The first CANDU Pressurized Heavy Water Reactor (PHWR) in Korea, Wolsong Unit 1 Nuclear Power Plant (WS1-NPP), has completed refurbishment. During the refurbishment of WS1-NPP, a significant amount of equipment and facilities were upgraded with regard to their safety aspects. In order to evaluate the effectiveness of WS1-NPP after these upgrades, new safety analyses were performed using the most recent technical standards for CANDU reactors concerning Design Basis Accidents (DBAs). The refurbished WS1-NPP is expected to be licensed for continuous operation based on the verified safety analysis results that were obtained using the upgraded computer codes and C-6 Rev. 1 of the newly adopted technical standards.

2. Technical Standards

2.1 Technical standards for safety analysis of WS1-NPP

Originally, the technical standards for the safety analysis of WS1-NPP were based on the single and dual failure criteria of AECB-1059 [1]. In order to verify its safety, WS1-NPP has adopted C-6 Rev. 1 [2] of the most recent technical standard for DBAs. For WS1-NPP, the Canadian Nuclear Safety Commission (CNSC) regulatory documents AECB-1059, R-10 [3], and C-6 Rev. 1 are the safety analysis standards, and R-7 [4], R-8 [5], and R-9 [6] are the design requirements for the special safety systems. These documents are selectively applicable, since they may not satisfy the criteria in some specific areas due to their publication being later than the WS1-NPP construction.
Among the existing CANDU reactors, the C-6 Rev. 1 has not yet been applied to the safety analysis. During the license of WS1-NPP, it was found that Canadian utilities (e.g. Point Lepreau, Gentilly-2 NPPs) chose some accidents from the C-6 Rev. 1 and then announced the technical standard as C-6 Rev. 1. WS1-NPP positively reviewed the Canadian utilities approach and decided to follow their approach; they analyzed the document for additional analysis items after benchmarking. Before proceeding, C-6 Rev. 1 and Rev. 0 [7] were compared and significant differences were not found, only the accident combination methodologies and radiation dose limits were different, as shown in Table 1. The combination of the accidents is ambiguous in C-6 Rev. 1 and the required dose limits are not practical because there are no rules at the present time. For the consistency with WS234-NPP, the combination of accidents and dose limits were maintained using the WS234-NPP methodologies. The WS234-NPP events were grouped according to the expected frequency from most likely occurrence (Class 1) to least likely occurrence (Class 5). The appropriate C-6 Rev. 0 class for an event was usually selected considering the expected frequency of the event. C-6 Rev. 0 has no frequency guidelines, so Ontario Hydro and CNSC agreed on a relationship between the frequency and class for the Darlington NPP, as shown in Table 2. It is the same as C-6 Rev. 1, and WS1-NPP also used the same methodology.

Table 1. Radiation dose limits for accident conditions.
a. From CNSC document C-6 Rev. 0.

Class	Individual dose limit (mSv)	
	Whole body	Thyroid
1	0.5	5
2	5	50
3	30	300
4	100	1000
5	250	2500

b. From CNSC document R-10.

	Single failures		Dual failures	
	Whole body	Thyroid	Whole body	Thyroid
Individual	5 mSv	30 mSv	250 mSv	2500 mSv
Population	100 man-Sv		10,000 man-Sv	

Table 2. Relationship between frequency and C-6 Rev. 0 event class

C-6 Event Class	Frequency Range (per reactor year)
1	$10^{-2} \leq f < 1$
2	$10^{-3} \leq f < 10^{-2}$
3	$10^{-4} \leq f < 10^{-3}$
4	$10^{-5} \leq f < 10^{-4}$
5	$f < 10^{-5}$

2.2 Results of the Canadian utilities safety analysis benchmarking for WS1-NPP

The CNSC required the most recent standard (C-6 Rev. 1) for refurbished plants for the safety analysis. However, the Point Lepreau and Gentilly-2 NPPs performed reviews and submitted their original licensing approaches as single and dual failures of AECB-1059, with the addition of only some accidents.

This was done in order to maintain consistency with their safety reports. The CNSC approved their approaches.

After the Canadian utilities safety analysis benchmarking, it was found that there were three categories of the additional analysis [8, 9]. Table 3 is taken from C-6 Rev. 1, Table 4 from the results based on the Canadian utilities and CNSC negotiation, and Table 5 from PSA for Severe Accidents Management. A loss of the shield cooling initiator was added in addition to the five initiators analyzed for the Canadian utilities, because the Core Damage Factor (CDF) caused by the loss of the shield cooling removes approximately 15% of the total CDFs in severe core damage at WS1-NPP. Table 3 indicates that most accidents were evaluated in WS1-NPP, except the ECC conditioning signal and dose analysis for GAI 95G02. The required items were analyzed and issued to the regulatory body at the end of 2011.

Table 3. Terms included for additional events of C-6 Rev. 1 of the Canadian utilities safety analysis.

WS1-NPP Chapter	WS1-NPP	Canadian utilities
15.2.1.8	Multiple steam generator tubes rupture	Multiple steam generator tubes rupture
15.2.4.A.4.2.2	Spurious opening of a liquid relief valve	Spurious opening of a liquid relief valve
15.3.2.A.4.5	Loss of feedwater flow to one boiler	Loss of feedwater flow to one boiler
15.4	Moderator system failures	Moderator system failures
15.5	Shield cooling failure events	Shield cooling failure events
59RF-AR-58	Shutdown cooling (SDCS) events	Shutdown cooling (SDCS) events
15.2.1.3.D	Submit ('11.12)	ECC conditioning signal & dose analysis for 95G02

Table 4. Additional analysis events based on Canadian utilities and CNSC negotiation.

No.	Canadian utilities	WS1-NPP
A-1	Review of C-6 Rev.1 events summary assessment	Submit ('11.12) in another report
A-2	Post-accident main control room operator dose	FSAR 6.9
A-3	Moderator subcooling improvement	Retubing
A-4	Review of analysis related to trip coverage with fresh fuel for secondary side events	FSAR 15.3
A-5	Summary of analysis for events in which moderator tritium can be released	FSAR 15.4
A-6	Overview of PSA in support of closing GAI 95G02	FSAR 15.2.1.3.D
A-7	Impact of safety analysis in support of refurbishment on plant operation	FSAR 15 General

Table 5. Additional events of PSA for Severe Accidents Management for Canadian utilities.

No.	Canadian utilities	WS1-NPP
P-1	Small LOCA and stagnation feeder break scenarios	These accidents are not DBA, Submit ('11.12) in another report
P-2	Station blackout scenarios	
P-3	Shutdown state scenarios	
P-4	Steam generator tube rupture scenarios.	
P-5	LOCA with failure of ECC and moderator	
	No analysis	Loss of shield cooling

3. Conclusion

The safety analysis for the refurbished WS1-NPP was performed according to the C-6 Rev. 1 of the most recent technical standard and C-6 Rev. 0 of the WS234-NPP safety analyses. This approach to the safety analysis of WS1-NPP is based on the safety analysis for WS234-NPP at the same site. The benchmarking results of the safety analysis approach of the Canadian utilities were also reflected in the development of this approach. The results of the safety analyses were in good agreement with the accepted criteria. The refurbished W1-NPP is expected to be licensed for continuous operation based on the verified safety analysis results that were obtained by applying the new technical standards. Among the existing CANDU, C-6 Rev. 1 was not fully applied to the safety analysis in the Canadian utilities. The CNSC required the most recent standards (C-6 Rev. 1) for the refurbished plants to be used in the safety analysis, although the Point Lepreau and Gentilly-2 NPPs undertook original licensing approaches for the single and dual failure of AECB-1059. Compared with the Canadian utilities, WS1-NPP has used the most recent safety standards,

REFERENCES

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[2] "Safety Analysis of CANDU Nuclear Power Plants", Draft Regulatory Guide C-006 Rev. 1 [E], Issued for public comments by the Atomic Energy Control Board, September 1999 [3] "The Use of Two Shutdown Systems in Reactors", R-10, AECB Regulatory Policy Statement, January 11, 1977.

[4] "Requirements for Containment Systems for CANDU Nuclear Power Plants", R-7, AECB Regulatory Policy Statement, February 21, 1991.

[5] "Requirements for Shutdown Systems for CANDU Nuclear Power Plants", R-8, AECB Regulatory Policy Statement, February 21, 1991.

[6] "Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants", R-9, AECB Regulatory Policy Statement, February 21, 1991.

[7] "Requirements for the Safety Analysis of CANDU Nuclear Power Plant", AECB Consultative Document C-6 Rev. 0, June 1980.

[8] Presentation to ANRA staff, "Deterministic Safety Analysis in Support of PLGS Refurbishment", October 2008.

[9] H. M. Huynh for Meeting with KHNP, KEPRI & KINS in Montreal, "G2 Refurbishment Licensing Framework", February 2006.