Technical Standards for Wolsong Unit 1 Nuclear Power Plant

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1. Introduction

More than twenty years after commencing commercial operation in 1983, Wolsong Unit 1(W1-NPP), the first CANDU Pressurized Heavy Water Reactor (PHWR) in Korea, has been undergoing refurbishment. Safety analyses were required to evaluate the safety of W1-NPP because significant amount of equipment has been refurbished. To evaluate the effectiveness of W1-NPP after these upgrades, new safety analyses were performed using the same technical standards of Wolsong Units 2, 3, 4 (W234-NPP) for Design Basis Accidents (DBA). The refurbished W1-NPP is expected to be licensed for full power operation based on the verified safety analysis results that are obtained by using the upgraded computer codes and newly adopted technical standards of W234-NPP.

2. Technical standards

2.1 Technical standards for W1-NPP

Originally, W1-NPP based on single & dual failure criteria of AECB-1059. To improve the safety of W1-NPP and to maintain the same technical standards for safety analysis at the same site, W1-NPP has newly adopted the same technical standards as those used in W234-NPP for design basis accidents. For W1-NPP, the Canadian Nuclear Safety Commission (CNSC) regulatory documents AECB-1059[1], R-10[2] and C-6 Rev.0 are safety analysis standards and R-7[3], R-8[4], and R-9[5] are design requirements for the special safety systems. These documents are selectively applicable, since they may not satisfy criteria in some specific areas due to their publication being later than the W1-NPP construction. For most events, as a basis for the licensing safety analysis, individual dose limits are specified in CNSC consultative document C-6 Rev.0 [6] and in Regulatory Document R-10. In such cases, both limits must be satisfied and the criteria shown in Table 1 for each event are the minimum of the two limits. The events are grouped according to expected frequency from most likely occurrence (Class 1) to least likely occurrence (Class 5). The appropriate C-6 Rev. 0 class for an event is usually selected by considering the expected frequency of the event. C-6 Rev.0 has no frequency guidelines, so Ontario Hydro and the CNSC agreed on a relationship between frequency and class for Darlington, as shown in Table 2. These technical standards are applied to W1-NPP in the same manner as they were in Darlington and in WS234-NPP.

Table 1. Radiation dose guidelines for accident conditions
a. From CNSC document C-6

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 Whole body	Single failures Thyroid	Dual failures Whole body	Dual failures 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 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 Result of frequency analysis for W1-NPP

Before applying C-6 Rev.0, we reviewed the link between C-6 and the event cases of the WS1-NPP safety analysis. All the applicable C-6 Rev.0 event case items that have been identified for W1-NPP are listed in Table 3. Some W1-NPP events have been reclassified using frequency based criteria, as was done in W234-NPP. After reviewing the PSA frequency results, Table 4 was made to show that only eleven cases are changed, as were the cases before in W234-NPP. Only two cases are needed to change to the severe class. But, for consistency, W1-NPP maintains the same event class as that used in W234-NPP.

Table 3. Summary items of compliance for C-6 Rev.0

	CL 1	CL 2	CL 3	CL 4	CL5	Systematic review	NDB	C-6 4.10 clause
C-6	13	10	9	13	128*	-	9	-
W1	13	10	9	13	118	23	9	2
W234	13	10	9	13	118	24	9	2

* End Fitting/Lattice Tube Failure was not considered at CANDU-6
NDB : Non Design Basis Accident

Table 4. Result of frequency analysis for W1-NPP

C-6 Rev.0 event	W1-NPP Frequency	W1-NPP Class	W234-NPP Frequency	W234-NPP Class
Turbine−generator load rejection	$f < 0.1$	1	$10^{-2} < f$	1
Flow blockage in any single reactor fuel channel assembly plus − failure of emergency coolant injection	$f < 1.2 \times 10^{-4}$	4	4×10^{-5}	5
Failure at any pipe location in the system that controls the pressure and inventory in the reactor main coolant system plus – degraded operation of containment atmosphere cooling equipment	$f < 10^{-5}$	5	$f < 10^{-6}$	5
Loss of class IV plus − failure of single S/G tube	$2.2X10^{-3}$	3	$f < 10^{-5}$	5
Loss of class IV plus − failure of Large LOCA	$f < 2 \times 10^{-5}$	5	$f < 10^{-5}$	5
Flow blockage in any single reactor fuel channel assembly plus total failure of containment atmosphere cooling equipment: both doors of the airlock or transfer chamber open most critical for the release of radioactive material from containment	$f < 2 \times 10^{-5}$	NDB	1.3×10^{-6}	NDB
End fitting failure plus total failure of containment atmosphere cooling equipment: both doors of the airlock or transfer chamber open most critical for the release of radioactive material from containment	$f < 2 \times 10^{-5}$	NDB	1.1×10^{-5}	NDB
End fitting failure plus total failure of containment atmosphere cooling equipment: both doors of the airlock or transfer chamber open most critical for the release of radioactive material from containment	$f < 2 \times 10^{-5}$	NDB	$10^{-7} < f \leq 10^{-6}$	NDB
Pressure tube/calandria tube failure plus total failure of containment atmosphere: cooling equipment both doors open of the airlock or transfer chamber most critical for the release of radioactive material from containment	$f < 2 \times 10^{-5}$	NDB	$10^{-6} < f \leq 10^{-5}$	NDB
Pressure tube/calandria tube failure plus total failure of containment atmosphere cooling equipment: both doors of the airlock or transfer chamber open most critical for the release of radioactive material from containment	$f < 2 \times 10^{-5}$	NDB	$f < 10^{-6}$	NDB
Reactor main coolant system large LOCA plus total failure of containment atmosphere cooling equipment: both doors of the airlock or transfer chamber open most critical for the release of radioactive material from containment	$f < 2 \times 10^{-6}$	NDB	2×10^{-6}	NDB

2.3 Technical standards for CANDU-6 in CANADA

CNSC required the most recent standards (C-6 Rev. 1) for the refurbished plants for the safety analysis. But Point Lepreau and Gentilly-2 plants performed reviews and submitted their technical standards [7], [8] while maintaining single & dual failure of AECB-1059; some of the additional accidents were added to the Point Lepreau safety analysis. This was done to maintain consistency with their Safety Reports. CNSC approved their approaches.

The review of accident cases for W1-NPP was compared to the additional safety analysis of the Point Lepreau plant. The list of results is shown in Table 5, indicating that all accidents were evaluated in W1-NPP except for ECC conditioning signal & dose analysis for GAI 95G02.

Table 5. Terms included for additional events of the Point Lepreau safety analysis

W1-NPP Chapter	W1-NPP	Point Lepreau
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
-	-	ECC conditioning signal & dose analysis for 95G02

3. Conclusion

The safety analysis for the refurbished W1-NPP was performed according to newly adopted technical standards and methodologies. This approach to the safety analysis of WS1-NPP is consistent with the safety analysis for WS234-NPP at the same site. The results of the safety analysis are in good accord with the acceptance criteria. The refurbished W1-NPP is expected to be licensed for full power operation based on the verified safety analysis results that were obtained by applying the new technical standards.

REFERENCES

[1] D.G.Hurst and F.C. Boyd, "Reactor Licensing and Safety Requirements", AECB-1059, June 1972.

[2] "The Use of Two Shutdown Systems in Reactors", R-10, AECB Regulatory Policy Statement, January 11, 1977.

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

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

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

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

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

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