# AN ASSESSMENT OF PRESSURE TUBE RUPTURE EVENT IN THE REFURBISHED WOLSONG-1 NUCLEAR POWER PLANT

**S.R. Kim, B.J. Moon, J.Y. Huh, C.J. Choi and D.S. Lee**
NSSS, Korea Power Engineering Company, Inc.

**S.M. Kim and Y.S. Lee**
NETEC, Korea Hydro & Nuclear Power Company, Inc.

## Abstract

A postulated pressure tube rupture (PTR) event, which is one of the in-core loss-of-coolant accidents (LOCA), has been analysed and evaluated for the refurbished Wolsong-1 Nuclear Power Plant (NPP). This plant is planned to undergo a long period of maintenance shutdown, beginning from the late 2009. The major activities would be the replacement of all 380 fuel channels, calandria tube assemblies and the connecting feeder pipes. As a part of its refurbishment project, a full scope of safety analyses is being jointly performed by Korean engineering companies. The industry standard toolset (IST) codes developed by CANDU Owners Group and updated models including ageing parameters are applied to the event analysis. The developed methodology and the results of pressure tube rupture event assessment are presented herein. The analysis results show that the implementation of the developed ageing model with the newly introduced computer codes is proved to be successful in the Refurbished Wolsong-1 NPP.

## 1. Introduction

Wolsong-1 NPP is scheduled to undergo a long maintenance outage starting from the late 2009. The major activities would be the replacement of all 380 fuel channels, calandria tube assemblies and the connecting feeder pipes. Ever since Wolsong-2, -3 & -4 projects were completed in the late 1990s, Korea has had no chance to apply state-of-the-art CANDU technologies which were instrumental in developing IST-code version, referring to Canadian Nuclear Safety Commission (CNSC) General Action Items (GAIs) and updated models for safety analyses. The Korean team is trying to apply IST codes and some of CNSC GAIs to the refurbished Wolsong-1 NPP.

The main changes of CATHENA Mod-3.5d/Rev.02 [1] enhance the application of ageing parameters in the circuit model of the heat transport system (HTS). In the Wolsong-2, -3 & -4 safety analyses [2], most thermal-hydraulic analyses of upset conditions have been performed with single-average-channel models of the four core passes. The 95 channels in a core pass are averaged such that the model contains 95 identical channels. Such a model may be unable to predict the multi-channel effects on the overall primary heat transport system response. This effect results in higher inlet header temperatures, reduced flows in some channels, and higher void fraction in channel outlets. The thermal-hydraulic behavior in postulated accident scenarios can also be affected. Therefore, a thermal-hydraulic model of the Wolsong-1 NPP is required to include ageing effects.

One of the postulated in-core LOCA is PTR event. The assessment of the fuel integrity is required for PTR accident. The thermal hydraulic analysis of ageing model was introduced to determine the effect of trip parameters, to evaluate the fuel integrity, and to assess shutdown system number 1 (SDS1) reactivity depth for the refurbished Wolsong-1 NPP.

In this paper, the developed methodology and the results of PTR event are presented and compared with the results of Wolsong-2, -3 & -4 NPP described in the final safety analysis reports (FSAR) issued in 1996.

## 2. Analysis methodology

### 2.1 Analysis method

The PTR event analysis process is shown in Figure 1. The thermal-hydraulic response of heat transport system is analysed by the computer code CATHENA. The coolant discharge rate and enthalpy obtained from CATHENA are used as input data for the moderator analysis and the evaluation of the fuel channel integrity. The results of moderator analysis are used as input for the containment analysis and the radioactivity calculations. The average coolant density and fuel temperature determined from CATHENA are provided as input data for the RFSP code to evaluate SDS1 depth margin calculation.

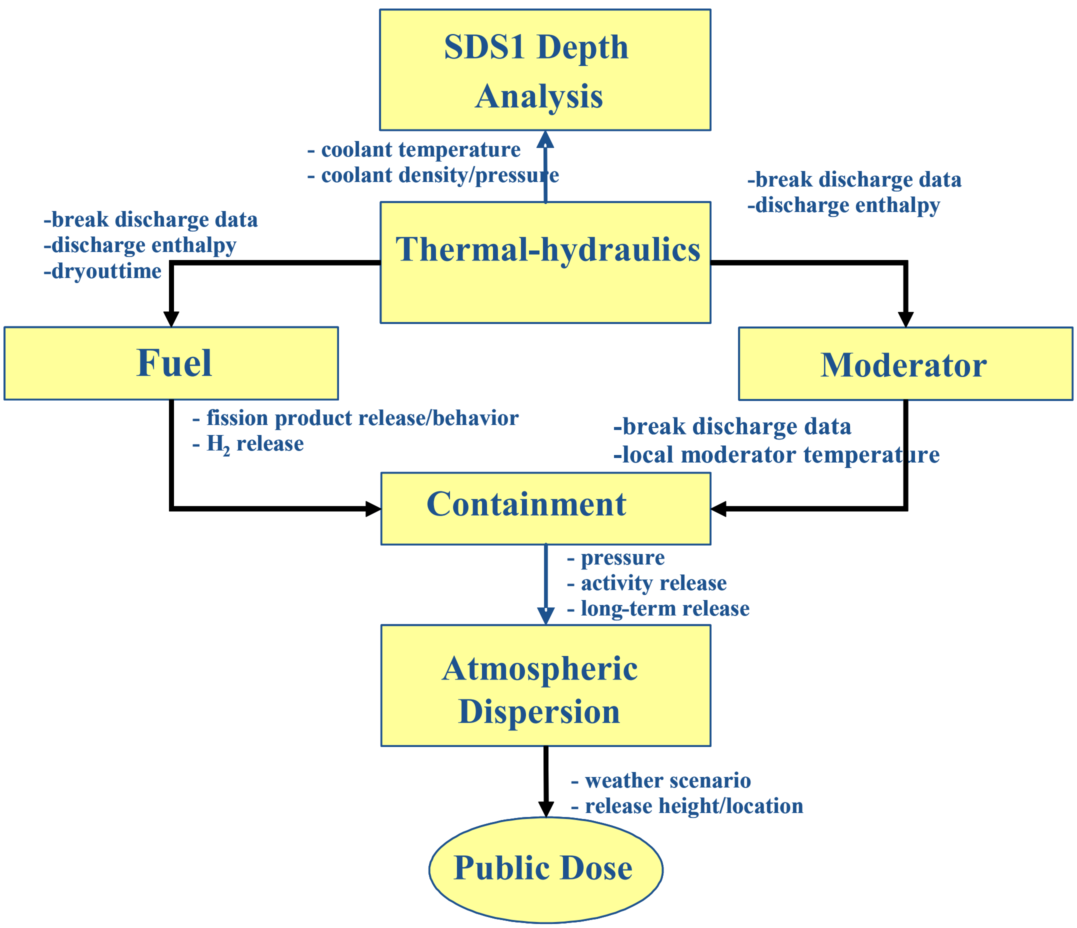



Figure 1 PTR Analysis Process

### 2.2 Thermal-hydraulic analysis model

#### 2.2.1 System model

The thermal-hydraulic analysis is performed with the CATHENA MOD 3.5d/Rev.2 computer code. The initial reactor power is assumed to be at 103% power to account for bulk power uncertainties.

The initial reactor inlet header (RIH) temperature may have an effect on the thermal-hydraulic behavior. The amount of void in the core increases with RIH temperatures, while the circuit flow decreases. The quality can affect the timing of the relevant process parameters for the PTR. The RIH temperature variations are considered in order to:

- Maximize the potential for an early dry-out: The initial RIH temperature is set as close to 267$^{o}$C for an aged core conditions at 103% power.
- Maximize the discharge rate: The lower RIH temperature of 261$^{o}$C is assumed to cover off the effect of possible steam generator (SG) tube cleaning which may occur during the operational life time after refurbishment.

#### 2.2.2 Multi-channel averaged circuit model

The heat transport system model includes a multiple average channel representation of the CANDU reactor core [3]. In this representation, each core pass is made up of 7 groups as shown in Figure 3 (i.e., 95 channels per core pass are distributed among 7 groups). The channel groupings are based on channel power and elevation. In each core pass, channel groups 1 through 4 contain high power channels from the inner region of the core; groups 5 to 7 contain lower power channels from the periphery of the core. Most of the high power channels have 6 MW power or higher (at full power), and having no orifice in the inlet feeder. The lower channels have power less than 6 MW at full power and a flow-reducing orifice in the inlet feeder. The channel power and axial power distribution from physics analysis are used as fuel power distribution for thermal-hydraulic analysis. Core pass 4 is represented by 7 averaged channels (representing 94 channels) in parallel with a single channel which is shown in Figure 3 (the broken channel).

#### 2.2.3 Single channel model

Channel O6_mod has the same geometry with O6 channel but the channel power and the bundle power of the two center bundles are modified to have the licensing limits of 7.3 MW and 935 kW, respectively. O6_mod is selected to be the broken channel to maximize the rate of moderator temperature increase.

#### 2.2.4 Break model

A guillotine break of pressure tube is assumed, and it is modeled by disconnecting the normal flow link and implementing an artificial valve model. An example of the break nodalization is shown in Figure 4. The break-developing time is assumed to be 0.01 second, and the two-phase discharge coefficient ($C_D$) is assumed to be unity. The calandria tube is assumed to fail and the fuel is ejected into the calandria vessel thus causing to maximize the coolant discharge rate into the moderator.

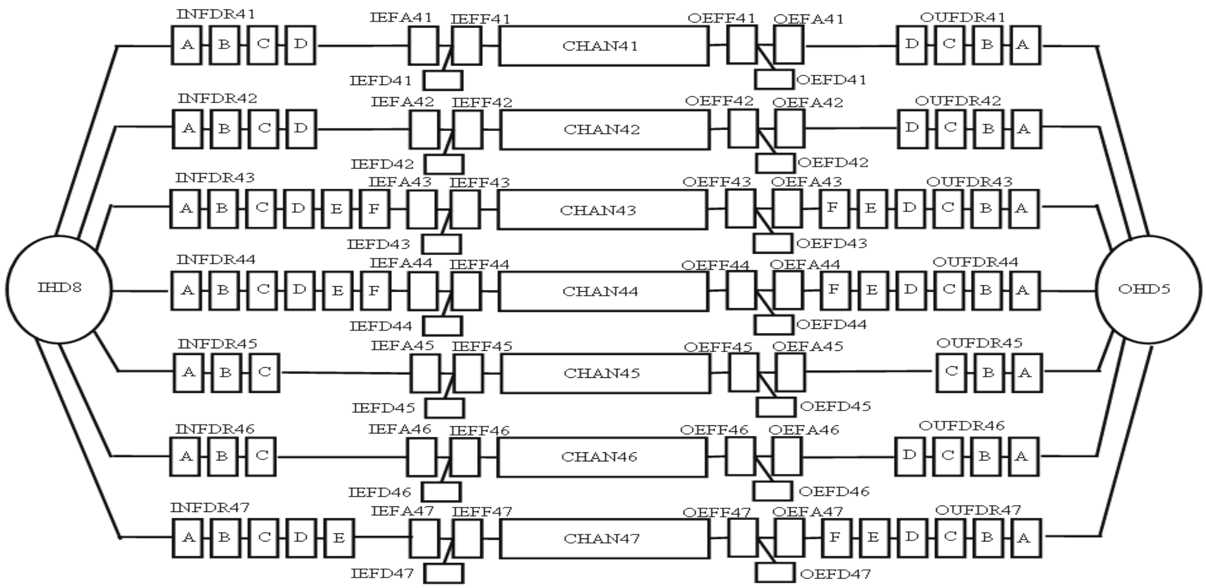



Figure 2 Multi-channel averaged circuit model (core pass 4)

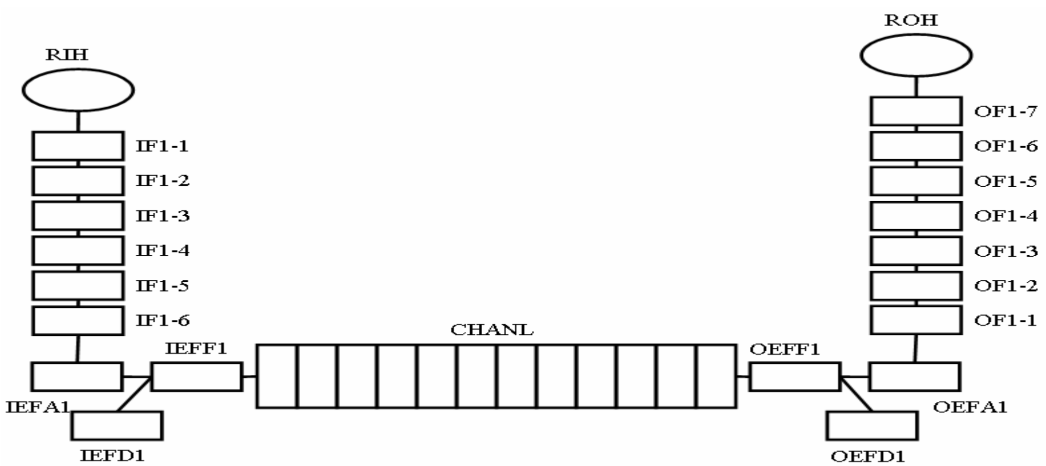



Figure 3 Single channel O6 model

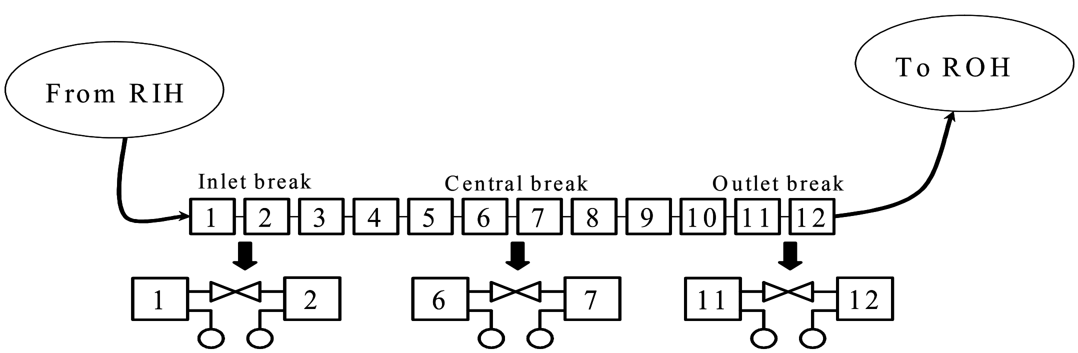



Figure 4 Break location and model nodalization of channel O6

### 2.3 Ageing model

The most important ageing effects are pressure tube diametral creep, steam generator tube fouling, feeder roughening, and feeder orifice degradation. The plant ageing model is based on the report in Reference 4 with creeping profile updated after the refurbishment of Wolsong-1 NPP. The creep profile for each averaging channel group is generated using RC-1980 equation and increased by 6.2%. In the single channel analysis of high-power channels for the aged core, a maximum diametral creep of 3.8% is considered [4]. The main effects of ageing are as follows:

- The onset of sheath dryout tends to occur earlier due to the significant critical channel flux value reduction with crept channels, and
- The inlet header temperatures tend to be higher due to reduced heat transfer effect in the boilers associated with boiler tube fouling, which tends to increase the void fractions in downstream channels.
- Hydraulic ageing effects may slightly alter the pressure distribution around the heat transport system, and the flow distribution among the channels. For example, low-power channels may have slightly higher flows due to feeder orifice degradation.

## 3. Analysis results

### 3.1 Trip coverage

The maximum mass discharge rate from the inlet side of break to the moderator is predicted to be about 250 kg/s, i.e., the initial discharge rates for a guillotine break in channel O6_mod with all the 12 fuel bundles are assumed to be discharged to the moderator. This discharge rate is slightly larger than a 1.5% RIH break, when comparing the initial maximum discharges. As considering ageing parameters, the effective trip parameters exist more than those of Wolsong-2, -3 & -4. Therefore, there are two effective trips on all power regions for SDS1 and SDS2 shown in Figures 5a, -5b, -5c and -5d, respectively.

### 3.2 Assessment of break core conditions and locations

First, the circuit simulation with single parallel channel model O6_mod produces the highest break discharges (in terms of energy discharge) for both refurbished core and aged core and then for the three channel locations. Figures 6 and 7 show the discharge power (MW) for the core conditions and three break locations along channel O6_mod for the first 500 seconds of the transient. As shown in Figures 6 and 7, the discharges are very similar for all three locations and core conditions. These are expected since the fuel is assumed to be ejected from the channel at the beginning of the transient. Discharge rate to moderator is affected slightly due to increased feeder pipe roughness, pressure tube creep, and initial RIH temperature. This has a slight impact on reactivity insertion, which would occur due to poison dilution. The initial core is more limited due to higher level in moderator, which gives a much larger effect. Therefore, the SDS1 depth analysis is performed with the initial core conditions but the long-term analysis is preformed with the break at inlet side of the channel of the aged core conditions.

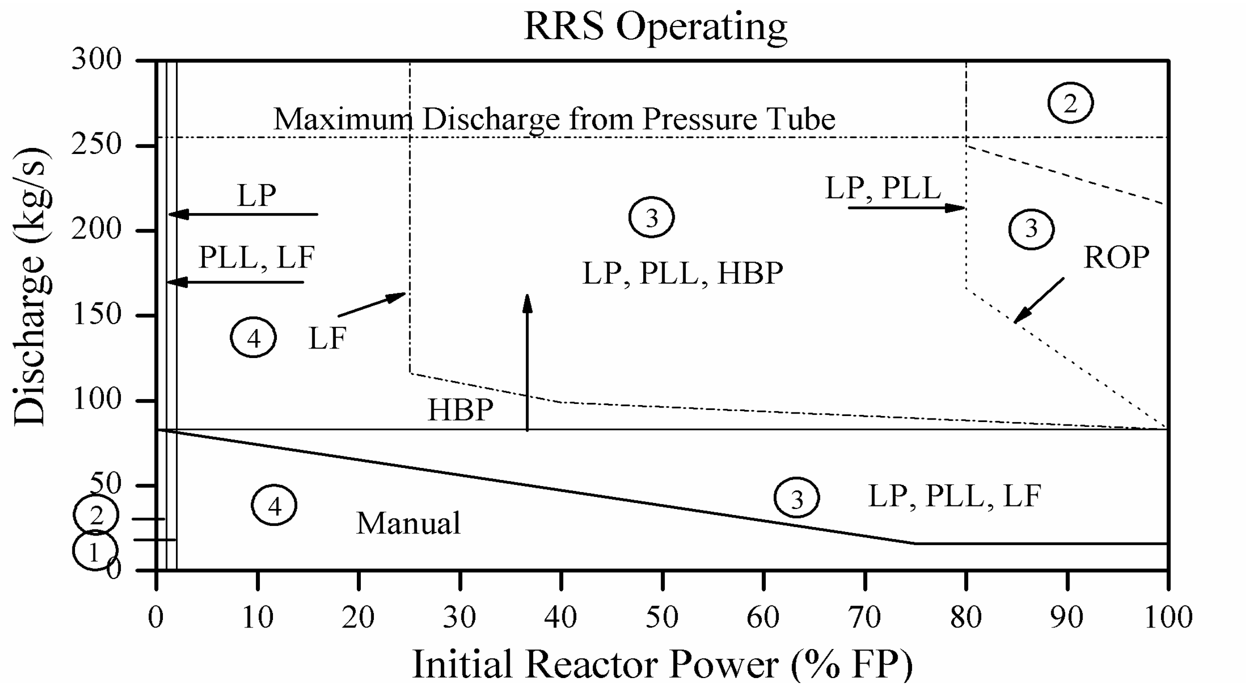



Figure 5a Trip coverage map for SDS1 (RRS operating)

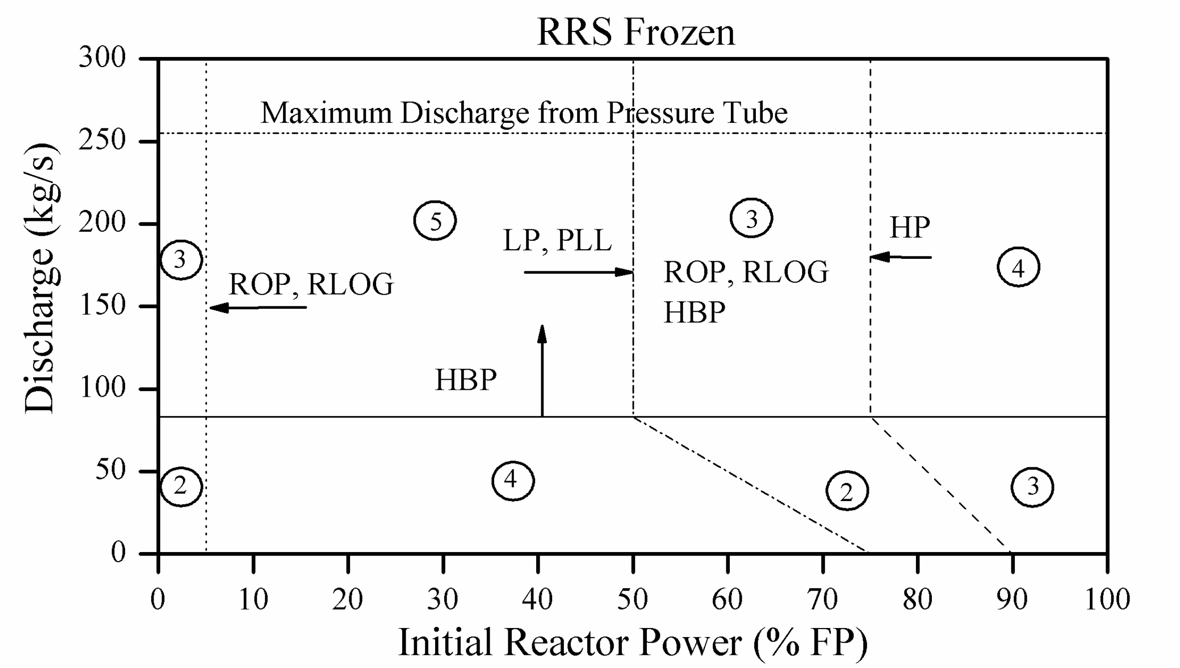



Figure 5b Trip coverage map for SDS1 (RRS frozen)

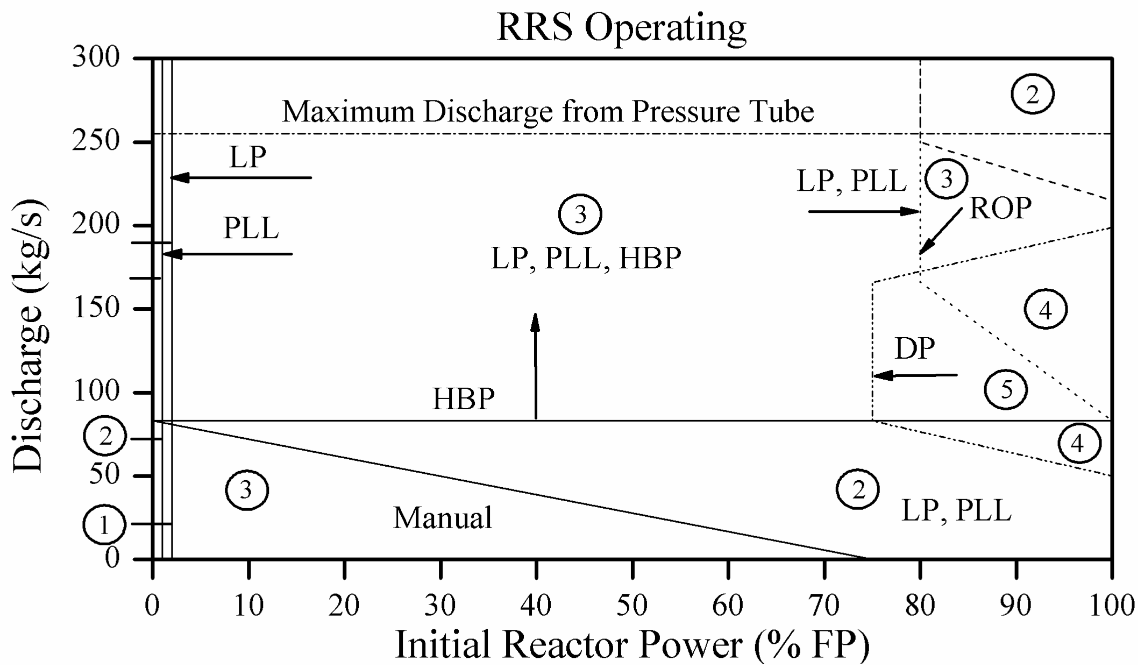
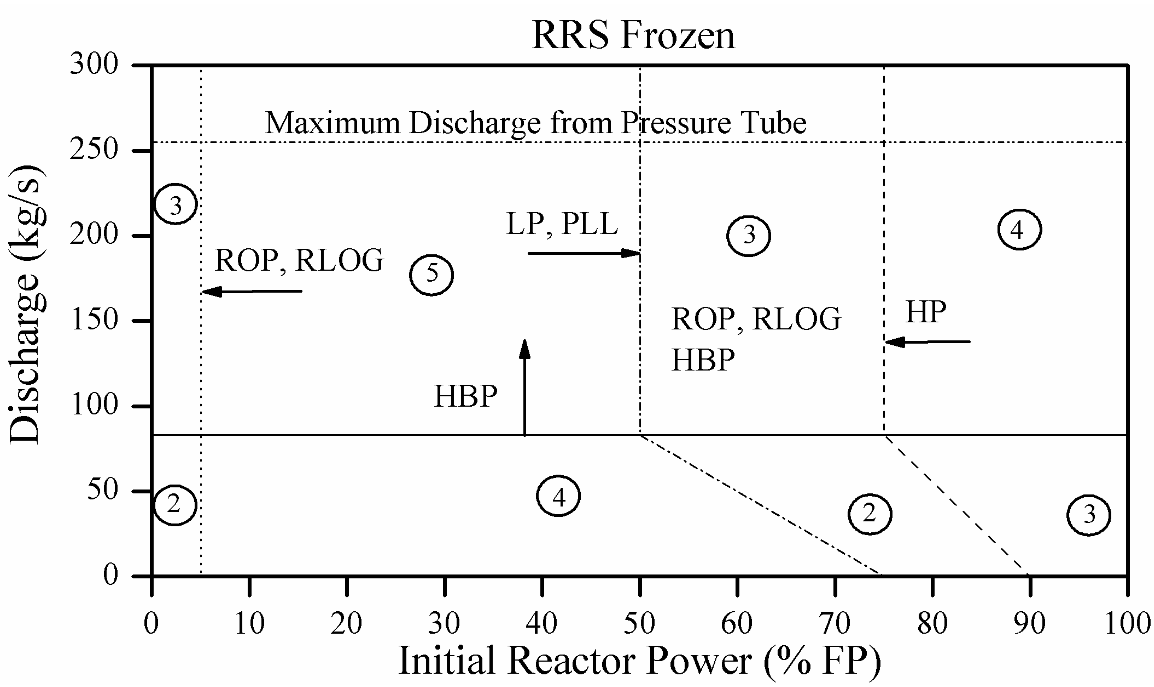



Figure 5c  Trip coverage map for SDS2 (RRS operating)




Where,

| | | | |
|---|---|---|---|
| HP | High HTS Pressure Trip | LP | Low HTS Pressure Trip |
| PLL | Pressurizer Low Level Trip | ROP | High Neutron Power Trip |
| RLOG | High Log Rate Neutron Trip | LF | Low HTS Flow Trip |
| HBP | High Reactor Building Pressure Trip | N | Number of Effective Trips |

Figure 5d  Trip coverage map for SDS2 (RRS working)

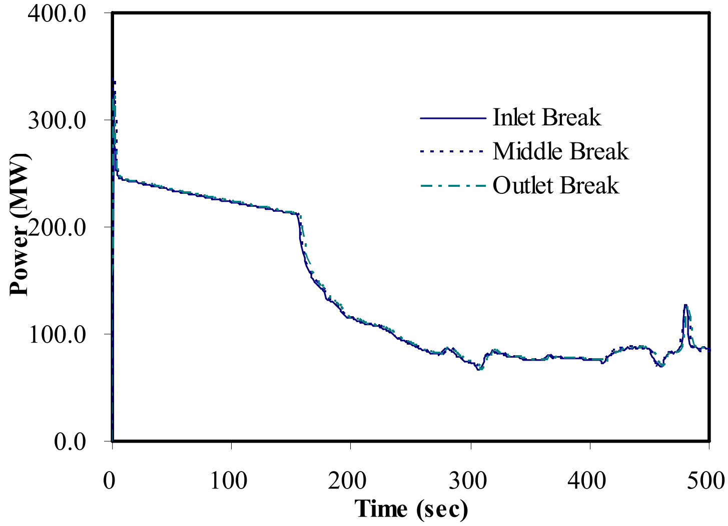



Figure 6 Total energy discharge (break locations)

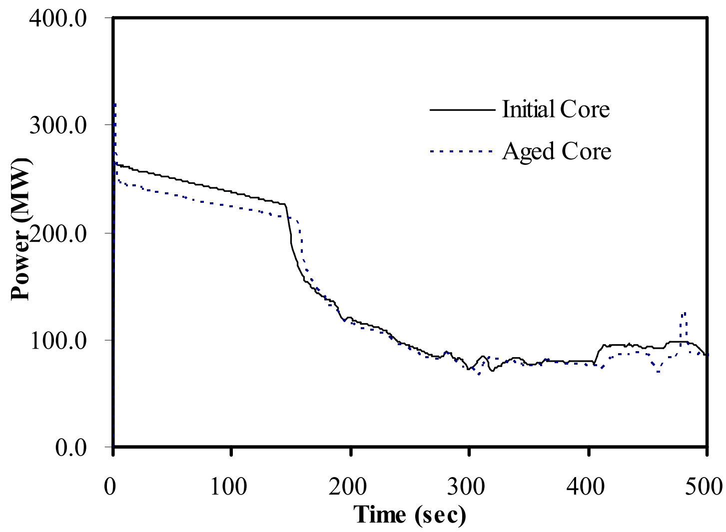



Figure 7 Total energy discharge (core conditions)

### 3.3 Heat transport circuit simulations

The compared initial heat transport system conditions between the Refurbished Wolsong-1 and the Wolsong-2, -3 & -4 for the aged core are given in Table 1. The RIH pressure is lower and the core flow is higher compared with Wolsong-2, -3 & -4. The higher flow predicted by the Wolsong-1 model is considered more accurate. The flow, listed in Table 1, is higher in the Wolsong-1 model due to the lower RIH temperature, the removal of large loss coefficients in the SG tube pipe models, the modeling of STM-GEN-COND in the SG tube pipe models, and the modeling of pressure tube creep in the channels, which increases flow area. The pressure drop across the core is lower (thus lower RIH pressure) due to the modeling of pressure tube creep. Above the headers, the pressure drop is increased due to fewer boiler tubes, higher flow, and increased boiler tube roughness. Differences in the primary heat transport system pump data and average feeder modeling methodology may also contribute to minor differences in predicted steady state flow and pressure distribution.

Table 1 Initial conditions for average channel circuit model (103% power)

| Parameters | Pass 1 ~ 3 | | Pass 4 | | | |
|---|---|---|---|---|---|---|
| | W-1 | W-2, -3 & -4 | Averaged from 94 channels | | Channel O6_mod | |
| | | | W-1 | W-234 | W-1 | W-234 |
| RIH pressure [MPa(a)] | 11.2 | 11.4 | 11.1 | 11.4 | 11.2 | 11.4 |
| RIH temperature ($^oC$) | 267 | 268 | 267 | 268 | 267 | 268 |
| ROH quality | 2.8 | 4.8 | 2.87 | 4.8 | 2.87 | 4.8 |
| Core flow (kg/s) | 2033 | 1900 | 2012 | 1874 | 24.3 | 22.6 |
| HT pump diff. P (MPa) | 1.8 | 1.8 | 1.78 | 1.8 | 1.78 | 1.8 |

As listed in Table 2, reactor trip (the first signal; low pressurizer level, the second signal; low HTS pressure trip) by second trip signal occurs earlier for PTR event compared to that for Wolsong-2, -3 & -4. The Wolsong-1 model includes a tracking algorithm to account for the mass of $D_2O$ feed pumped into the PHTS from the $D_2O$ storage tank. Once the mass of $D_2O$ storage tank water used corresponds to the low alarm level setpoint, the $D_2O$ feed is assumed to stop. This then allows the PHTS pressure to decrease more quickly in the Wolsong-1 simulation, resulting in earlier trips on the low pressurizer level and the low HTS pressure. Also, the lower RIH temperature in the Wolsong 1 model means that the initial pressurizer level is lower and therefore is closer to the trip setpoint at the start of the accident transient. As the HTS cools down following reactor trip, emergency core cooling system (ECCS) is initiated to make up PHT system inventory and PHT pump tripped when PHT pressure reaches the setpoint of 2.6 MPa(a).

### 3.4 Single channel simulation

Transient header conditions generated for the aged circuit simulations with single parallel channels are applied as boundary conditions of the O6_mod channel using CATHENA MOD3.5d/Rev.2. Throughout the transient, the fuel sheath and pressure tube are well cooled at temperatures below 363$^o$C and 313$^o$C, respectively.

## 4. Conclusion and discussion

For the pressure tube rupture, the trip coverage analysis is assessed based on the small LOCA analysis results, by comparing discharges with the developed CATHENA trip coverage model [5]. Adequate trip coverage is demonstrated for this event. Also, the results have reasonable trip parameters when comparing with the previous results of Wolsong-2, -3 & -4.

Table 2 HTS and ECCS event sequence for PTR

| Event | Circuit O6_mod Time (s) | |
|---|---|---|
| | W-1 | W-2, -3 & -4 |
| Reactor trip | 154 | 199 |
| LOCA signal | 187 | 231 |
| Crash cool-down | 217 | 261 |
| Start of HP ECC injection | 268 | 322 |
| PHT pump trip | 407 | 455 |
| Start of MP ECC | 952 | 1413 |
| Start of LP ECC | 4264 | 3232 |

The circuit and single channel simulation results indicate an adequate cooling of the intact channels in both the intact loop and the broken loop for both the refurbished and the aged core. The maximum sheath and pressure tube temperature are predicted to be well below the criterion for the aged core conditions.

Since commencing the commercial operation, various components in the main Wolsong-1 heat transport system have experienced deterioration. We have, therefore, drawn up the ageing model capable of modeling the ageing parameters and capturing important individual feeder characteristics such as elevation, which incorporates some individual channel characteristics. These effects produce higher inlet header temperatures, reduced flows in some channels, and higher void fraction in channel outlets than the initial core conditions of Wolsong-2, -3 & -4.

It has been also confirmed that the developed ageing model and the newly introduced computer codes have successfully tested the pressure tube rupture event in the Refurbished Wolsong-1.

## 5. References

[1] B.N. Hanna, Editor, "CATHENA MOD-3.5d/Rev 2 Input Reference", 153-112020-UM-002, Revision 0.0, 2005.

[2] KHNP, "Final Safety Analysis Reports for Wolsong-2, -3 & -4 NPPs ", 1996.

[3] B.N. Hanna, Editor, "CATHENA MOD-3.5d/Rev. 2 Theory Manual", 153-112020-UM-001", Revision 0.0, 2005.

[4] AECL, "The Development of the Wolsong-1 Aged NUCIRC Model at 11 EFPY after Retubing", 2008.

[5] S.R. Kim, etc., "CATHENA Circuit Trip Coverage Model", 59RF-03500-AR-002, 2008.