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AGING EFFECT ON THE FUEL BEHAVIORS FOR CANDU FUEL SAFETY ANALYSIS

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ABSTRACT – Because of the aging of heat transport system components, the reactor thermal-hydraulic conditions can vary, which may affect the safety response. In a recent safety analysis for the refurbished Wolsong 1 NPP, various aging effects were incorporated into the hydraulic models of the components in the primary heat transport system (PHTS) for conservatism. The aging data of the thermal-hydraulic components for an 11 EFPY of Wolsong 1 were derived based on the site operation data and were modified to the appropriate input data for the thermal-hydraulic code for a safety analysis of a postulated accident. This paper deals with the aging effect of the PHTS of the CANDU reactor on the fuel performance during normal operation and transient period following a postulated accident such as a feeder stagnation break.

Introduction

A recent safety analysis for the refurbished Wolsong 1 considered the aging effect of the reactor by assuming that the core is in an 11 EFPY (effective full power year) state. Here, an 11 EFPY core means that the core was assumed to be operated for 11 EFPY after the refurbishment of Wolsong 1. From the site operation data and thermal-hydraulic analysis results, the aging data for major components of the reactor and heat transport system were derived. The derived data include the roughness data of the feeder pipe, steam generator, end-fitting and pressure tube, creep data for pressure tube, orifice degradation, and steam generator fouling, and so on [1]. The reason that the safety analysis was carried out for the aged core is just to obtain the conservative results in terms of the safety. However, a detailed comparison of the safety analysis results for the fresh and aged core has not been assessed for a specific postulated accident scenario.

In this study, a thermal-hydraulic simulation and a fuel behavior analysis were carried out for both fresh and 11 EFPY aged cores to compare the aging effect of the reactor components on the CANDU safety response. The analysis was conducted for a stagnation feeder break accident, which is one of the single channel accidents when the other channels remain intact in the CANDU core. A feeder break can occurr in any of 380 channels in the reactor at any time during the reactor's operating life. Because of this, a feeder break is assumed to occur in the high-powered 'limiting' channel, the power of which is a limiting operational condition of 7.3 MW for a conservative assessment.

From the thermal-hydraulic analysis of a circuit and one channel by the CATHENA [2] code, the critical break size of the inlet feeder pipe which causes the most stagnation flow in the channel was determined for each fresh and aged core. Fuel analyses were carried out for two steps: the first analysis evaluated the fission product inventory and its distribution for the normal operating condition using the ELESTRES-IST [3] code, and the second step assessed the fission gas release after the feeder break for each critical break size of the fresh and aged cores by applying Gehl's model [4].

Through the analysis results, the aging effect on the fuel safety was assessed, and it was also checked how much the aged core assumption was conservative for a stagnation feeder break accident.

1. Thermal-hydraulic Analysis for Fresh and Aged Core

Aging of the PHTS component of the CANDU reactor may cause some changes to the thermal-hydraulic characteristics, and these changes also affect the operational and safety margins of the CANDU reactor. The primary aging phenomena include pressure tube diametral creep, magnetite deposition, feeder orifice degradation, and fouling effect of the steam generator, etc.

To investigate the effect of component aging, thermal-hydraulic simulation was carried out for a postulated feeder stagnation break accident. A break in an inlet feeder can lead to a reduction in the coolant flow in the adjacent fuel channel with the channel remaining its power. Depending on the break size, a complete stagnation of channel flow can occur and result in rapid fuel and channel heat-up and channel failure. Since feeder breaks can be postulated to occur in any channel and at any time, thus, to ensure that the worst consequences are covered, a limiting channel was defined as a 7.3 MW channel.

The thermal-hydraulic analysis for the inlet feeder break was performed using the average channel circuit model with a specific broken channel for the CATHENA code [2]. The circuit model is used to determine the system response to a break size. The reactor inlet header conditions obtained from the circuit simulations were used as boundary conditions for analyzing the fuel and fuel channel behavior. A broken pass is represented by seven multiple average channels (94 averaged) in parallel with a single channel (the broken channel). The single channel analysis is performed to examine the channel thermal-hydraulic behavior. A single fuel channel model, which is for the limiting channel, covers a single channel from the inlet header to the outlet header only. Here, the limiting channel has the same geometry as channel O6 but the channel power and bundle power of the two center bundles have been modified to the licensing limits of 7.3 MW and 935 kW, respectively, and thus it is called the 'O6_mod' channel. The transient header boundary conditions of pressure, vapor enthalpy, liquid enthalpy, and void fraction generated from the circuit simulations were used in a single channel analysis.

1.1 Circuit Analysis Results

Thermal-hydraulic circuit analyses for each fresh and aged core were performed for various break sizes to determine the header boundary conditions in a stagnation state. Figures 1 and 2 show the mass flow rate results between the 6th bundle and 7th bundle in the broken channel for the aged core and fresh core, respectively. The most severe flow stagnations occurred at a break size of 18.0 cm^2 for the aged core, and at a break size of 17.5cm^2 for the fresh core. Thus, the header boundary conditions for these break sizes were provided to the input conditions for a single channel analysis.

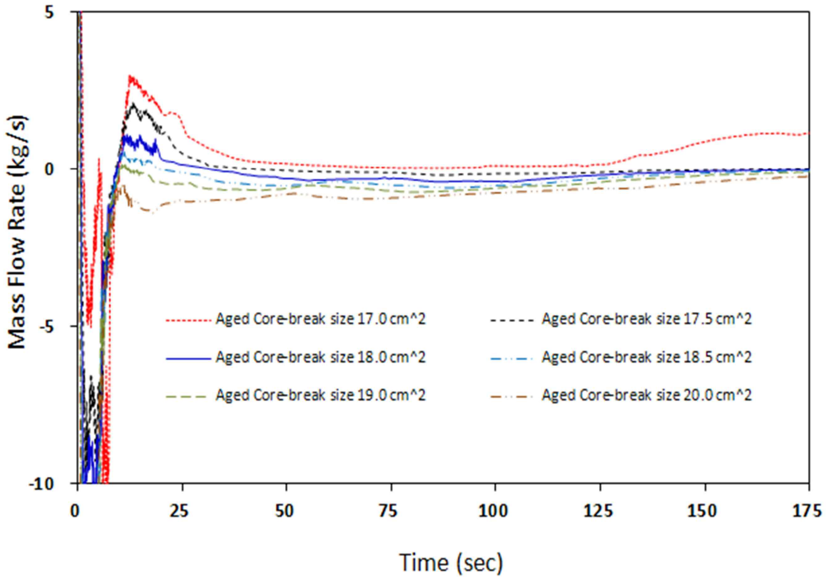

Figure 1 Circuit Analysis Results for Aged Core

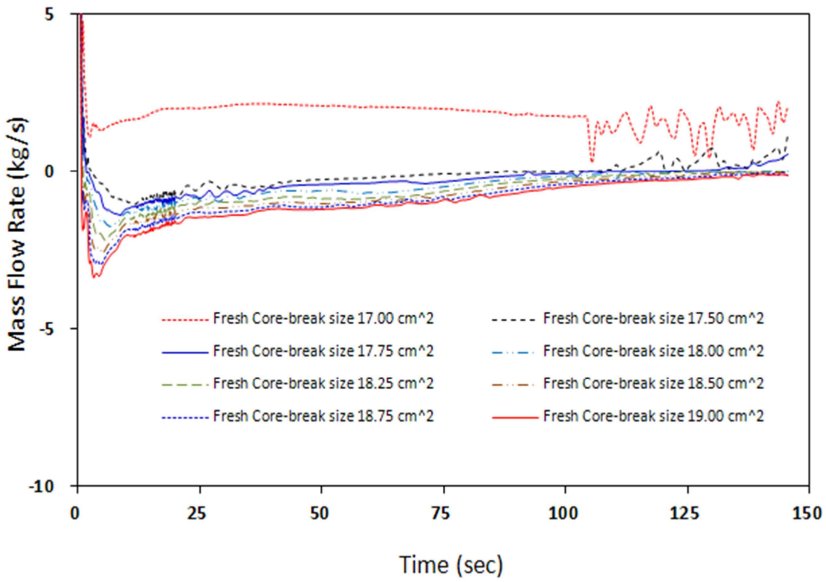

Figure 2 Circuit Analysis Results for Fresh Core

1.2 Single Channel Analysis Results

Stagnation feeder breaks result in channel failure owing to rapid fuel and channel heat up. Since the timing of the channel failure and the channel conditions at this time are a strong function of the break size, a parametric survey for the break size was therefore performed for each aged and fresh core.

Figure 3 shows the mass flow rate results for the aged core according to various break sizes at 10 seconds after the accident. The most severe flow stagnation was observed at a break size of 16.35cm^2. Table 1 summarizes the times and locations of the contact of the pressure tube and calandria tube (PT/CT) for various break sizes of the aged core. A pressure tube failure time of 11.37 seconds for the aged core was provided to the fuel analysis input to evaluate the fission product release.

Table 1 PT/CT Contact Times and Locations Following Break for Aged Core

Inlet Feeder Break Size (m^2)	Approximate Coolant Flow at 10 sec (kg/s)	Time of 1st PC/TC Contact (sec)	Location of 1st PT/CT Contact (BD position)
15.00×10^{-4}	2.9	31.07	12
15.50×10^{-4}	1.81	19.33	6
15.75×10^{-4}	1.39	15.31	6
16.00×10^{-4}	0.91	12.05	7
16.25×10^{-4}	0.35	10.08	7
16.35×10^{-4}	0.04	11.37	7
16.50×10^{-4}	-0.18	10.19	6
16.75×10^{-4}	-0.45	10.10	6
17.00×10^{-4}	-0.51	10.39	6
17.50×10^{-4}	-0.57	11.00	6
18.00×10^{-4}	-0.54	11.62	6
19.00×10^{-4}	-0.88	13.22	6
25.00×10^{-4}	-3.35	31.06	2

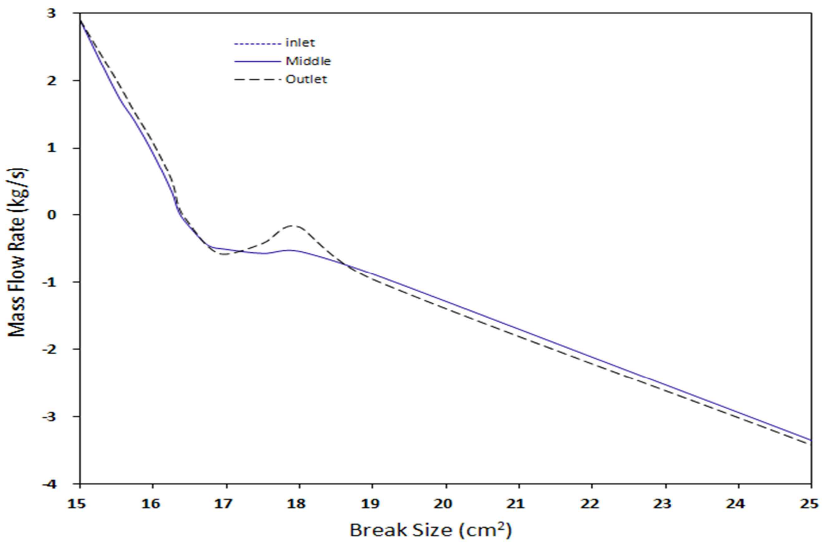

Figure 3 Aged Core Mass Flow Rate at 10 seconds after Accident for Break Sizes

Figure 4 shows the mass flow rate results for the fresh core according to various break sizes at 10 seconds after an accident. The most severe flow stagnation was observed at the break size of 17.50cm^2. Table 2 summarizes the times and locations of PT/CT contact for various break sizes of the fresh core. A pressure tube failure time of 10.34 seconds for the fresh core was provided to the fuel analysis input to evaluate the fission product release.

Table 2 PT/CT Contact Times and Locations Following Break for Fresh Core

Inlet Feeder Break Size (m^2)	Approximate Coolant Flow at 10 sec (kg/s)	Time of 1st PC/TC Contact (sec)	Location of 1st PT/CT Contact (BD position)
15.50×10^{-4}	3.02	75.63	12
16.00×10^{-4}	2.04	26.30	11
16.25×10^{-4}	1.85	20.86	11
16.35×10^{-4}	1.71	19.57	11
16.50×10^{-4}	1.62	18.08	11
16.75×10^{-4}	1.22	16.21	7
17.00×10^{-4}	0.83	12.72	7

17.25×10^{-4}	0.48	10.93	7
17.50×10^{-4}	-0.21	10.34	6
18.00×10^{-4}	-0.52	11.05	6
19.00×10^{-4}	-0.6	12.81	7
25.00×10^{-4}	-3.37	30.70	1

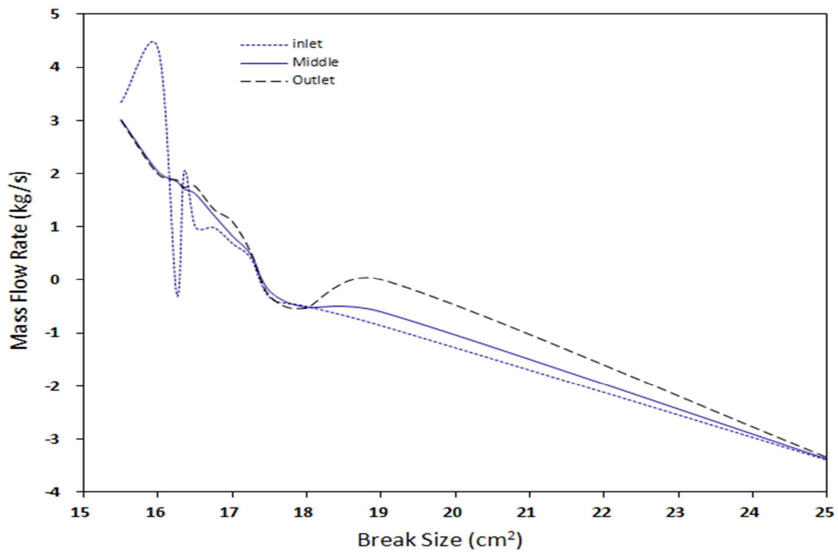

Figure 4 Fresh Core Mass Flow Rate at 10 seconds after Accident for Break Sizes

2. Fuel Analysis for Fresh and Aged Core

The objective of the fuel analysis for a feeder break accident is to estimate the quantity and timing of fission product release from the fuel in the affected channel. Fuel thermal and mechanical behavior depends on the coolant conditions, the power transient and the duration of the transient. A stagnation feeder break is characterized by rapid fuel and pressure tube heat-up and channel failure during a short transient period.

ELESTRES-IST [3] code was used in the fuel analysis to evaluate the fission product inventory and fuel behavior under normal operationg condition and provide the initial transient coolant conditions. Gehl's model [4] was used to assess the amount of fission gas released from the fuel during the transent period.

2.1 Fission Product Inventory during Normal Operation

Fission products are uniformly generated throughout the pellet at rates propotional to the local fission rate. However, fission poducts may move over time because of the diffusion, and changes in the granular structure of the UO_2. Gap fission products accumulated between the pellet and sheath can be promptly released following a sheath failure. Therefore, the amount of gap fission product is important because it may affect the initial dose to the public.

Table 3 shows the fission product results for the fresh and aged core for each isotope under normal operation. As shown in Table 3, the distribution of the fission product at a gap, grain boundary, and in-grain was different because the diffusion rate of the fission product is dependent on the fuel temperature.

Table 3 Fission Product Inventory for Each Isotope for Fresh and Aged Core

ISOTOPE	Gap (TBq)		Grain Boundary (TBq)		In-Grain (TBq)	
	Fresh	**Aged**	**Fresh**	**Aged**	**Fresh**	**Aged**
CS-137	6.9	7.4	26.5	27.0	161.7	160.8
CS-138	57.4	62.0	2272.1	2313.5	12833.8	12787.9
I-131	413.9	436.4	953.4	970.8	5023.3	4983.5
I-132	833.8	878.4	1462.3	1488.9	7498.6	7427.4
I-133	507.1	540.7	2259.7	2300.9	12389.3	12314.6
I-134	151.4	163.1	2602.4	2649.8	14642.4	14583.3
I-135	304.2	326.2	2128.3	2167.1	11837.2	11776.5
I-137	7.0	7.6	1258.6	1281.4	7076.0	7052.6
KR-83M	5.8	6.3	177.3	180.5	1004.2	1000.5
KR-85M	21.3	22.9	429.3	437.1	2426.8	2417.4
KR-85	0.6	0.7	3.2	3.2	19.4	19.3
KR-87	23.0	24.8	847.7	863.1	4801.4	4784.2
KR-88	46.9	50.6	1184.2	1205.7	6702.9	6677.7
KR-89	10.0	10.8	1708.7	1739.7	9606.7	9574.9
RU-103	441.5	466.0	1122.5	1142.7	5984.0	5939.2
RU-106	10.4	11.2	49.0	49.9	290.0	288.4
SR-89	651.4	686.3	1529.1	1556.5	8146.3	8084.0
SR-90	5.4	5.7	26.3	26.7	161.6	160.8
TE-131M	47.9	50.5	121.0	123.2	642.9	638.0
TE-131	74.2	79.8	903.7	920.2	5047.1	5025.1
TE-132	716.813	754.0	1438.9	1465.1	7494.3	7430.9
TE-133M	116.8	125.4	1005.7	1024.0	5601.5	5574.5

TE-133	85.4	92.0	1421.5	1447.3	7941.8	7909.3
TE-135	13.0	14.0	1277.5	1300.7	7176.4	7152.2
XE-133M	9.7	10.4	63.9	65.1	355.1	353.3
XE-133	865.2	920.6	2094.6	2132.72	11084.1	10990.5
XE-135M	4.9	5.3	390.4	397.5	2201.9	2194.4
XE-135	50.2	53.9	242.5	247.0	1333.6	1325.6
XE-137	14.8	16.0	2304.5	2346.3	12956.3	12913.3
XE-138	27.3	29.4	2237.1	2277.8	12614.2	12571.3

As shown in Figure 5, the difference in the gap fission product inventory between the aged and fresh core is higher at central bundles. Since the power of central bundles is higher than the other bundles, the fuel temperature of the aged core is higher than that of the fresh core, and this may affect the diffusion rate of the fission product from inside the fuel pellet to the gap. Fission product inventory at the grain-boundary shows similar results to that of the gap inventory as shown in Figure 6. When the fuel sheath fails after the postulated accident, the gap and grain-boundary inventory can be released promptly, and the aged core seems to be more conservative than the fresh core

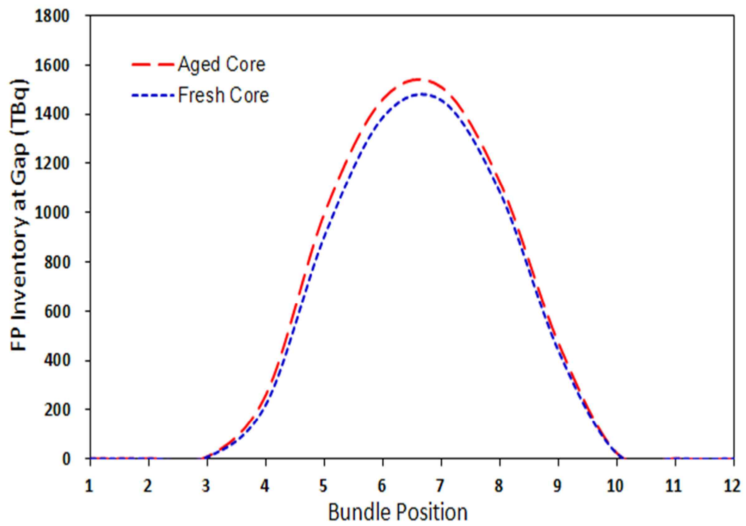

Figure 5 Fission Product Inventory at Gap for the Fresh and Aged Core

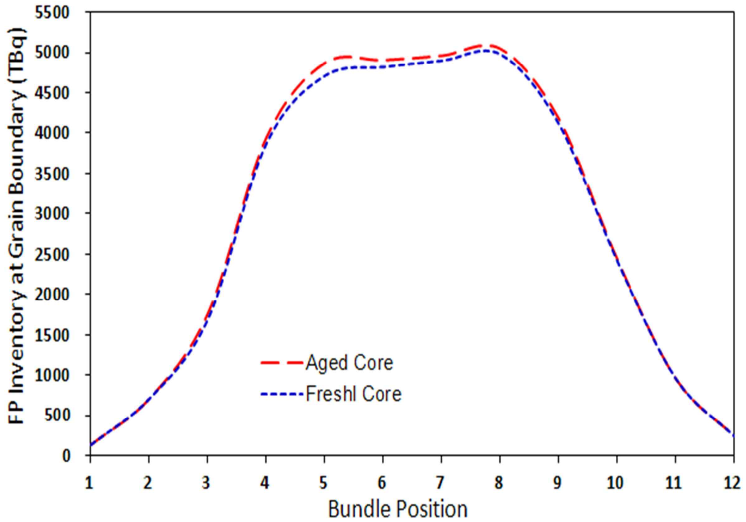

Figure 6 Fission Product Inventory at Grain Boundary for the Fresh and Aged Core

2.2 Fission Gas Release during Transient

To evaluate the amount of fission gas release more conservatively, it was assumed that all fuel sheaths in the broken channel were failed and the entire gap inventory was released instantaneously at the beginning of the accident. An additional calculation of the transient fission gas release from the fuel grains and grain boundary following a feeder stagnation break was evaluated by applying Gehl's release model [4]. The amount of released fission gas was calculated until the time of a fuel channel failure which was determined from the fuel channel T/H analysis and an additional 2 seconds. The limiting fuel channel of the fresh core was predicted to fail at 10.34 seconds for a break size of 17.5 cm^2, and the aged core's limiting channel was predicted to fail at 11.37 seconds for a break size of 16.35 cm^2 following the break

A total of 30 isotopes, including I, Kr, and Xe, were considered in the analysis of the fission gas release. The analysis results indicate that the initial release of the aged core was higher than that of the fresh core because of the prompt release of more gap inventory of the aged core. Since the channel failure of the fresh core occurred about 1 second earlier than the aged core, the released fission gas abruptly increased at this time. However, it was evaluated that the final cumulative fission gas release for the aged core was about 10% higher than that of the fresh core. From this result, it was found that the 11 EFPY aged core is more conservative than the fresh core for the fuel safety perspective.

3. Conclusions

The aging effect of the main components of the CANDU primary heat transport system on the thermal-hadraulic characteristics and fission gas release was investigated. Through the thermal-hydraulic analysis for the fresh and 11 EFPY aged core, the most severe stagnation break sizes for the inlet feeder break and the channel failure time were determined and the coolant conditions were provided to the input data for a fuel analysis. Based on th thermal-hydraulic data, the fission product inventory under normal operating condition was calculated for both fresh and aged core and the fission gas release was evaluated during the transient. The gap and grain boundary fission product inventories during normal operation for the 11 EFPY aged core were higher than those of the fresh core. In addition, it was also calculated that the final cumulative fission gas release for the aged core was about 10% higher than that of the fresh core following the feeder stagnation break. From this result, it was concluded that the 11 EFPY aged core is more conservative than the fresh core from a fuel safety perspective.

4. References

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Acknowledgement

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