Development of Steady-state Thermal-hydraulic model for Aged Wolsong-1

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1. Introduction

All industrial plants undergo changes with time and nuclear plants are no exception. Wolsong-1 started its commercial operation in 1983. Through 20 years of operation the plant has experienced aging behavior in many aspects. Specifically, aging in primary heat transport system, caused by CANDU-6's inherent design, can affect ability of fuel cooling. This may reduce the margin to Onset of Intermittent Dryout(OID), similar concept to Departure from Nucleate Boiling in Pressurized Light Water Reactor. To assess the current condition of primary heat transport system component, steady-state thermal-hydraulic model is developed.

2. CANDU-6 reactor system description

Wolsong-1 is a CANDU 6 Pressurized Heavy Water Reactor(PHWR). It has 380 horizontal fuel channels surrounded by a cool low pressure heavy water moderator. Each fuel channel is six meters long and contains twelve fuel bundles within a pressure tube. A bundle is made up of 37 elements which contain natural uranium in the form of compacted sintered cylindrical pellets of uranium dioxide (UO_2). Each channel has an End Fitting at each end, which allows for the Fueling Machines to attach to facilitate on power refueling. Coolant enters the channel from an inlet feeder pipe which is connected to the inlet end fitting. The coolant then enters the fuel string, flowing within the subchannels between the fuel elements inside the pressure tube. The coolant leaves the channel via the outlet feeder pipe which is attached to the outlet end fitting. The coolant enters the channel at about 11 MPa and 265 ^{o}C. It leaves slightly above 10 MPa and 310 ^{o}C.

The core is subdivided into two symmetrically located figure of eight loops. Each loop consists of two core passes of 95 channels each. Each pass contains a pump which feeds an inlet header which is connected to 95 inlet feeders which connect to the channels as described above. The 95 outlet feeders connect to an outlet header. Two riser pipes connect the outlet header to the hot leg side of a steam generator. Coolant flows through the vertical steam generator U-tubes to the steam generator outlet where it flows to the pump suction line of the pump in the other core pass in the loop. Because it is a figure of eight loop, flow in channels in one core pass is in the opposite direction to the other. The two Heat Transport System(HTS) loops are connected at each end of the reactor through the pressurizer interconnect line and the purification and feed interconnect lines. The pressurizer is connected to the discharge pipes of outlet headers 3 and 7. The purification feed flow is associated with inlet headers 2 and 6, while the purification return flow enters the HTS at the suction of pumps 1 and 3. The HTS also contains stability pipes which connect outlet headers 1 and 3 in loop 1, and outlet headers 5 and 7 in loop 2(See Figure 1).

3. Aging phenomena

Increasing of pressure tube diameter due to irradiation creep reduces the hydraulic resistance in the channel, hence increases flows, but causes the coolant to preferentially bypass the interior subchannels of the bundle, reducing Critical Channel Power(CCP). Because there is more creep in the higher power channels, there is a flow redistribution effect whereby some of the flow from the outer low power channels is redirected to inner channels. This mitigates the effect of pressure tube diametral creep on CCP for the central, ROP most limiting channels.

Hydraulic resistance is increased due to redistribution of magnetite in the HTS. Dissolution of iron and flow accelerated corrosion (FAC) is occurring in the outlet feeders. Iron is being removed from the outlet feeders and being re-deposited in the cold part of the circuit, including the cold leg of the steam generators, the inlet feeders, and possibly the first section of the channel. The magnetite layers cause both a fouling of the inside of the steam generator tubes, leading to reduced heat transfer, and also an increase in hydraulic resistance in the steam generator tubes and inlet feeders. This has a negative effect on core flow, on inlet header temperature and, consequently, on CCP.[1]

4. Thermal-hydraulic model

NUCIRC is a steady-state thermal-hydraulic code used by designers and analysts to examine the behavior of the HTS of a CANDU nuclear reactor over a wide range of single-phase and two-phase operating conditions. NUCIRC employs a forward marching iterative method to solve the conservation of mass and energy equations for steady, incompressible flow of Newtonian fluids.[2]

Plant operating data at around 80%FP are logged to develop single phase thermal-hydraulic model and data at 100%FP are gathered to confirm the ability of model to simulate single or two phase operating condition. The single phase data can be acquired during stat-up after outage. According to the operating procedure, power run-up ceased for 8 hours after reaching 80%FP to test. Data logged before power runs up again will be used for modeling. Operating data at full power was gathered a few days after plant being stable. Single phase operating data acquired on June 8, 2004(6577EFPD) and two phase operating data on June 16(6585EFPD) are used for modeling of Wolsong-1 PHTS.

Geometry data of HTS come from CANDU-6 generic design data. To consider pressure tube diametral expansion, creep rate on the position of each bundle of all channels should be prepared. Wolsong-1 has measured pressure tube diameter of some channels. With this measurement data and creep rate prediction program, provided by AECL, creep rate data of Wolsong-1's own is obtained.

NUCIRC, modularized computer code, can consider each sub-system independently and integrally. Modeling starts from below header model, purification system model and steam generator model and merge them into one integrated model. The integrated model includes fuel channel, feeder, header, steam generator, pump and interconnecting pipes. Information from sub-system model is used to make integrated full circuit model and fine tuning is needed.

5. Simulation results

The most representative parameters of HTS condition are temperature of reactor inlet header(T_{RIH}), pressure of reactor outlet header(P_{ROH}) and differential pressure between inlet and outlet header($\Delta P_{Header\text{-}to\text{-}Header}$). Thus, those parameters are the measure of assessing the practicability of the model. Table 1 shows the simulation results of single-phase and Table 2 shows those of two-phase. The Wolsong-1 thermal-hydraulic model predicts well the single-phase with an error of less 0.2%. The two-phase operating condition is simulated by an error of less 1.5%.

6. Conclusion

This thermal-hydraulic model for Wolsong-1 simulates the current operating condition of data acquisition point accurately and can be used as CCP analysis model to calculate Regional Overpower Protection(ROP) trip set point of Wolsong-1 reflecting the aging effects of pressure tube diametral creep and flow accelerated corrosion.

References

[1] W.J. Hartmann and M. Cormier, "Plant Aging Adjustments to Maintain Reactor Power at The Point Lepreau Generating Station", The 5th International Conference on CANDU Maintenance, 2000 Nov.

[2] D.J. Wallace, "NUCIRC Program Abstract and Theory Manual", TTR-765 Rev. 2, AECL, 2003 June.

Table 1 Simulation results of single-phase

Parameter	Site Data	NUCIRC Data	Error(%)
PROH(MPa(a))			
Pass23	9.948	9.948	0.00%
Pass41	9.966	9.973	0.07%
Pass67	9.940	9.940	0.00%
Pass85	9.990	9.995	0.05%
Δ P H-H(kPa)			
Pass23	1259.55	1259.03	-0.04%
Pass41	1221.88	1221.53	-0.03%
Pass67	1229.64	1227.66	-0.16%
Pass85	1224.50	1225.77	0.10%
TRIH(℃)			
Pass23	261.92	261.89	-0.01%
Pass41	261.89	261.85	-0.02%
Pass67	261.43	261.00	-0.17%
Pass85	261.45	261.50	0.02%

Table 2 Simulation results of two-phase

Parameter	Site Data	NUCIRC Data	Error(%)
PROH(MPa(a))			
Pass23	9.946	9.946	0%
Pass41	9.966	9.955	-0.11%
Pass67	9.932	9.932	0.0%
Pass85	9.985	9.968	-0.17%
Δ P H-H(kPa)			
Pass23	1281.14	1263.78	-1.36%
Pass41	1235.35	1241.28	0.48%
Pass67	1250.86	1234.18	-1.33%
Pass85	1237.16	1245.63	0.68%
TRIH(℃)			
Pass23	262.65	262.08	-0.22%
Pass41	262.49	262.76	0.10%
Pass67	262.10	261.75	-0.13%
Pass85	262.04	261.62	-0.16%

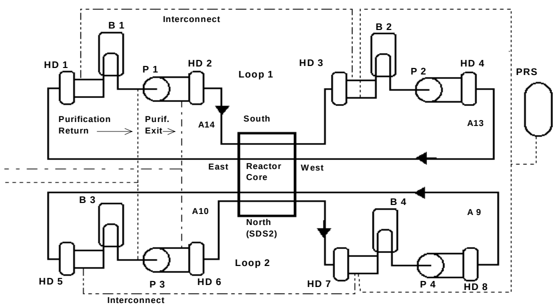

Figure 1 Primary Heat Transport System