Analysis of Moderator System Failure Accidents by Using New Method for Wolsong-1 CANDU 6 Reactor

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1. Introduction

The safety analysis required for the licensing application to restart the Wolsong-1 NPP after the major refurbishment has been carried out by using the most up-to-date code suite, namely, Canadian Industry Standard Toolset (IST) codes. Although the most of the safety analyses was carried out by using the IST codes, the moderator system failure accident analysis was performed by using the non-IST codes, namely, SMOKIN G-2 [1] and MODSTBOIL [2].

The SMOKIN-G2 calculates modal amplitudes which are used to reconstruct the neutron flux and the power generated in all areas of the reactor core. Whereas, the RFSP-IST [3] calculates neutron flux and power distributions in the reactor core based upon two energy-group, three dimensional neutron diffusion equations. To reconfirm the safety of moderator system failure accidents, the safety analysis by using the reactor physics code, RFSP-IST, coupled with the thermalhydraulics code, CATHENA [4] is performed additionally.

In the present paper, the newly developed analysis method is briefly described and the results obtained from the moderator system failure accident simulations for Wolsong-1 CANDU 6 reactor by using the new method are summarized.

2. Methods and Results

The moderator system failure accidents, such as, loss of heat sink and inventory of the moderator system are analyzed with the coupling method of CATHENA and RFSP-IST.

2.1 CATHENA and RFSP-IST Model

The model to simulate the thermalhydraulic behavior of the moderator system during transients is set up by using the CATHENA code [1], which is the state-of-the-art computer program for the thermalhydraulic modeling of CANDU-6 plants, in replacement of the previously used MODSTBOIL program [2]. In contrast to the former code, the later code does not represent the whole moderator system such as head tank, the moderator system piping.

Using CATHENA, these deficiencies are removed and the results so obtained are more reliable and more legitimate for the licensing purposes.

For the simulation of the reactor power transients that are coupled with the thermalhydraulic conditions during moderator system failure accidents, the core physics model set up by using the CERBRRS/CERBERUS modules of RFSP-IST [5] which contains the detailed and comprehensive algorithm of the CANDU 6 reactor regulating system(RRS).

The coupling between CATHENA and CERBRRS/CERBERUS is accomplished by using the moderator temperature, density and the moderator level drop calculated by CATHENA as input into the physics code. The reactor power calculated by CERBRRS/CERBERUS is then fed back into CATHENA as input to reflect the moderator system heat load. This cycle completes the coupling loop between reactor physics and thermalhydraulics and the transient simulations are advanced with the fixed time-step size of 0.5 seconds until the transients are terminated.

2.2 Analysis Results

2.2.1 Assumptions

The reactor core state is assumed to be at 100% FP after the restart from a long shutdown period. It is assumed that the moderator system failure accidents would not have an immediate impact on PHTS. Thus, the coolant temperature and density and the fuel temperature are remained unchanged during the transient to be the same as in the steady-state condition at time t=0 s before the accident starts.

2.2.2 Loss of Heat Sink

The loss of moderator heat sink takes place when either the service water to the moderator heat exchangers (LOSW) is lost or the moderator forced circulation is lost (LOMC). At the LOSW/LOMC events, the moderator heats up and swells, leading eventually to bursting of the rupture disks in the calandria relief ducts.

Following heat up and subsequent bursting of the calandria rupture disks, the net reactivity effect of moderator level, temperature and density results in the sub-criticality of the core system. In other words, once the D_2O inventory starts to be discharged through the rupture discs into the containment, the core system reactivity sharply drops even before the reactor trip, and the reactor operation enters into the self-shutdown mode.

The reactor self-shutdown phenomena are observed to be consistent with both the results of the analysis conducted by using the new method and the old method.

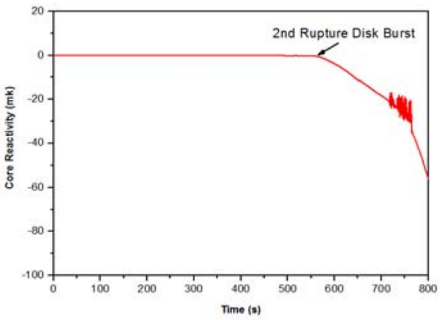

Fig 1. Core System Reactivity Transient Due To Loss of Heat Sink (No Reactor Trip)

2.2.2 Loss of Inventory

The moderator system pipe breaks cause the loss of moderator D_2O inventory. This event is characterized by the redistribution of reactor power, as the level drops the upper fuel channels are exposed and these channel powers are being reduced to decay power level, while the RRS responds by increasing power in the lower (bottom) part of the core in order to maintain the reactor total power in demand. Accordingly, the power distributions are appreciably changed (top-to-bottom tilt), which in turn could cause the onset of fuel sheath dryout in some fuel channels due to the localized power peak in the bottom region of core.

The slow loss of moderator D_2O inventory under the normal reactor operating condition at 100% full power level is analyzed for the purpose of assessing the engineering safety feature system capabilities and the reactor core passive features with and without the ROP trip credits. For this accident type, the simulation results show that the reactor operation also enters into the self-shutdown mode without the onset of fuel sheath dryout.

With the same conditions, the results of analysis by using MODSTBOIL/SMOKIN-G2 showed that the ROP trip of SDS1 and SDS2 occur when the moderator level approximately drops to the fuel channel row A. But in the present analysis, it is demonstrated that the reactor conditions at which this trip could be initiated do not occur. However, the results of the analyses by using the both methods show that the reactor shuts down naturally by itself without the intervention of any engineered shutdown device action due to the negative reactivity insertion induced by the moderator level drop.

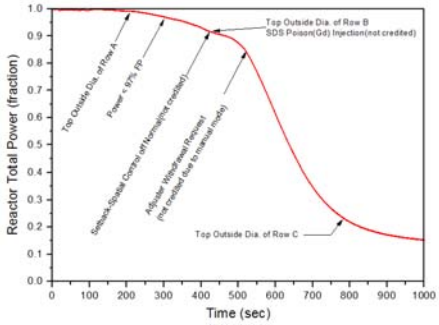

Fig. 2. Reactor Power Transient Due To Loss of Moderator D_2O Inventory with Leak Rate of 80 kg/s (No Reactor Trip).

3. Conclusions

The safety analysis of the moderator system failure accidents for Wolsong-1 CANDU 6 reactor was carried out by using the new code system, i.e., CATHENA and RFSP-IST, instead of the non-IST old codes, namely, SMOKIN G-2 and MODSTBOIL. The analysis results by using the new method revealed as same with the results by using the old method that the fuel integrity is warranted because the localized power peak remained well below the limits and, most importantly, the reactor operation enters into the self-shutdown mode due to the substantial loss of moderator D_2O inventory from the moderator system. In the analysis results obtained by using the old method, it was predicted that the ROP trip conditions occurred for the transient cases which are also studied in the present paper. But, in the new method, it was found that the ROP trip conditions did not occur.

Consequently, in the safety analysis performed additionally by using the new method, the safety of moderator system failure accidents was reassured. In the future, the new analysis method by using the IST codes instead of the non-IST old codes for the moderator system failure accidents is strongly recommended.

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