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Review of Different Models of Coastdown Transient in Pressurized Water Reactor

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ABSTRACT

Problems related to the loss of coolant flow in nuclear reactor may initiate fuel meltdown and fuel-cladding interaction (FCI) due to the overheating of fuel and are therefore of great concern in power reactor safety. Therefore, licensee must provide evidence through rigorous analyses of all conceivable flow problems that plant's engineered safety systems (EES) have the capability to maintain fuel and cladding temperatures well below the melting point. Among the loss of flow events the flow coastdown transient is also a critical issue considered for the safety analysis of Pressurized Water Reactor (PWR), which is characterized by a sudden loss of power to the main reactor coolant pump (RCP). There has been a provision in RCP to maintain flow through reactor core for some time, immediately after the loss of power to pump, like in case of Station Blackout (SBO) due to the flywheel mechanism of RCP. However, that is inadequate for the extended times following the SBO and RCP must be powered by emergency diesel generators (EDGs) to maintain flow through the reactor core to remove heat from the fuel without any break. After the event of Fukushima, a lot of progress has been made to analyze situations, where the EDGs become inundated or unavailable. Analytical and empirical models have continuously been evolved to simulate the characteristics of pumps in such a crucial event to guide accident prevention and mitigation strategies. These models are divided into two broad categories like the short models and detailed models. The short models take into account the inertia of the flywheel, pump speed and the flow rate in core. The detail models also consider the pump characteristic curve on which homologous curves are derived and help to establish head and flow rate third degree polynomial. It has been observed that the detail models predict more accurate results in comparison with the experimental data. It has also been observed that the accuracy of the simulated results also relies on the inclusion of the pump mechanical friction losses in the model. Moreover, an attempt has been made to extend the coastdown transient analysis to predict the core outlet temperature during the course of the accident which requires an efficient solution strategy for solving models for the pump, coolant half time and the core time constant. In this article, evolution of different models has been discussed in detail.

Keywords: *Coastdown, Energy Ratio, Inertia flywheel, Transient Flow*

1. Review of Different Models

The coastdown transient is a highly sensitive issue as the pump head is no longer available owing to the loss of the off-site power. As a result, the reactor is scrammed, nevertheless, the heat generation in fuel continues due to decaying fission products. Additionally, a serious setback can be faced in case the diesel generators are unable to restore the power supply for keeping the coolant pumps in operation. Quantitatively, the decay heat piles up and approximately within one hour and reaches to 1.5% of the steady state power before its shutdown. Eventually, the fuel assemblies may disintegrate or at least experience bowing, cladding ballooning, etc. which cannot be ignored from safety perspective. So, the safety analysis of the nuclear reactor imposes the stringent requirement of having enough cooling ability to keep the fuel temperature below the specified limit. So, a lot of effort has been made for studying the coastdown transient because the fate of the core depends upon the inertia of the flywheel of the pump. Basically, coastdown transient models can be classified in two main categories, termed as the short models and detailed models. The simulation with the short models needs only design head, rated flow and moment of inertia. While the detail models consider the pump characteristic curve upon homologous curves are drawn and pump head and flow rate were calculated by fitting the third order polynomial to these homologous curves.

Analytical methods were the first to be developed for the establishing the characteristics of pump during operational transients. In these methods, mathematical models of varying complexity considering characteristics of energy, torques, friction of pump and associated components in the flow loops were suggested. First analytical method to study the behavior of pump during operational transients utilized the model for hydraulic characteristics of machine and pipeline and projected them on a single diagram[1]. The model was generalized to incorporate start-up transients as well as valves opening and closing by improving energy conservation considerations [2]. They also linked the accuracy of their method to the availability of components' steady state input data for the flow prediction[2]. Then, a dynamic model by combining the pump rotary parts and fluid in the connecting pipes for the coastdown event originating from the failure of the power appeared, in which the resulting equations were solved analytically for the rigid fluid columns and the elastic fluid columns. Also, the technique was extendable to simulate other abnormalities in the system[3]. The calculation of An analytical method to determine velocity of flow during a coastdown transient in a loop without the use of pump characteristic curves the for single-suction and double-suction centrifugal pumps was suggested [4]. While comparing the predictions of analytical method and experimental characteristics' curves, it was found that the effect of a mechanical friction loss on the flow rate was very small in the early stage of pump failure transient, and the time of two-third decay of the flow was not affected very much by the friction loss. However, this effect is larger in the later stage of the flow decay. Therefore, the time when the flow rate becomes zero, depends very much on the estimation of this loss [5].

With respect to nuclear power plant, a model considering the torque speed dependence to electric torque to predict primary coolant flow rate and the primary pump rotation speed for different transients was suggested [6]. The method

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was applicable to simulate various scenarios like station blackout, pump startup and the valve opening installed on the pump discharge line.

Following the improving trend of envisioning of the accident scenario, a pragmatic model relying on the motor and the revolving parts of the pump was established [7]. However, model did not consider the electrical energy and therefore, no longer suitable for coastdown transient analysis. While analyzing the test data for the PWR at the Shipping port (Pa.) Power Station. (NSA 22: 33101), kinetic energy in pump rotor was found to be a significant parameter in determining the rate of flow during coastdown. Therefore, an empirical method for treating the transfer of energy from the pump rotor to the fluid was developed [8].

Another experimental validation study of a thermal-hydraulic model[9] revealed that in first half the simulated results were of higher side while the converse behavior was observed in the second half of the time-line of the coast down event. For having deep insight on the pump start-up related issues, the torque equation was solved adopting the detailed characteristic data of the pump[10]. A theoretical model [11] was developed by investigating the pump characteristic like flow rate, pump speed and rotational speed in a stopping period of the centrifugal pump experimentally. It was explored that, initially, the pressure rise coefficient is higher than the quasi-steady state coefficient and at the later stage the converse situation was observed. The main finding of this study was that the impulsive pressure and the lag in circulation formation around impeller vanes play predominant roles in the difference between dynamic and quasi-steady characteristics of turbo pumps. To address the safety related issues of nuclear power plant a simple method utilizing steady state governing equations of centrifugal pump characteristics was developed with the hypothesis that pump head is changing proportional to the square of the pump speed [12]. Despite the simplicity of his method, he successfully handled the highly fast startup issue of the pump and expanded its application to the pump stopping case. The method was successful as no big anomalies were reported in comparison with the experimental trends. Another experimental and theoretical study [13] on the stopping and starting periods of a centrifugal pump on a set-up having a small pipeline connected to discharge flange of the pump revealed that unsteady pump dynamic’s characteristics considerably deviates from steady-state characteristics. The researchers also solved their numerical model by the method of characteristics.

In another study, dedicated to characterize the experimental facility ATLAS (Advanced Thermal–hydraulic test Loop for Accident Simulation)with reference to scaling and behavior of the pumps in single and two phase flow cases came up with the homologous curves in single phase in all quadrants [14]. While characterizing KALIMER-600 against coastdown accident, various influential factors such as the size of the flywheels, initiation of decay heat and peak temperature were considered to optimize the design with a coastdown time of 25 s for the safe end of the accident [15]. Furthermore, the investigation of the coastdown transient for a pool type Material Test Reactor (MTR) was carried out employing a mathematical model for kinetic energy of pump and in the piping system [16]. The results predicted by simulation have shown an excellent agreement to that of the experiments results. The worthwhile effort regarding coastdown transient has been made for Dayawan nuclear reactor which was a three loops nuclear power plant, where a simplified mathematical model was developed for solving flow rate transient and pump speed transient during flow coastdown period [17]. The quadruplicate polynomial curve equation was found to adequately simulate the flow-rate, rotate speed along with time for the analysis of station blackout accident at nuclear main pump [18]. The simulated strategy depends upon the total pressure drop relation, balanced momentum equation of the primary coolant, coolant pump momentum balance relation, pump half time, coolant half time and the energy ratio. The energy ratio which was foundation stone of their study depends upon pump half time and coolant half time and directly affect all pump parameters, such as the speed, non-dimensional flow rate, torque and head of the pump. They solved the resulting equations analytically for finding the transient flow rate and pump speed. From the comparison to the experimental data, it can be inferred that the calculated flow rate values agree well with the experimental values in the first half of the flow coastdown process. In the later-half of the experiment, the calculated values were over-estimated in comparison with the experimental data. The well-known reason is that of ignoring the mechanical friction loss of the coolant pump as the mechanical friction loss becomes dominate when the pump speed approaches to zero.

Similarly, a mathematical model of core thermal-hydraulics for the flow of coastdown transient has been developed for power as well as in the research reactor and compared results to the experimental results[19]. They also simulated the core temperature with simple heat transfer relation instead of solving the complex equations as programmed in popular transient simulating code (RELAP). The solution technique of their model incorporates parameters like, pump half time, the loop half time and the core time constant. The excellent agreement between experimental and simulated core temperature has been observed. Recall that such simulation methods are only suitable for case when there is no boiling in the core. Initially, the matching trend between experimental and the simulated results was better and after that the trend becomes wider that might be due to ignoring the mechanical friction in pump. The other reason might be the complete stoppage of the one coolant pump during the course of the accident.

The parameters of the primary coolant pump in the Jordan Research and Training Reactor (JRTR)were studied by using a software package, Modular Modeling System (MMS) and characteristics have been developed [20]. The loop coolant inertia effect was found to be small in the JRTR PCS loop, i.e., about one second increases in a coastdown half time required to halve the coolant flow rate. The coastdown half time was found to linearly increase with the flywheel inertia, however remained within the safe limits of fuel integrity [21]. While studying the coastdown transient in the generation IV reactors ESPR–sodium cooled fast reactor, it was found that the Generation IV reactors have more safety as compared to their predecessors, as a tertiary loop has been added [22]. Briefly, the ESPR is as industrial type reactor having three cooling systems- primary secondary and tertiary systems. The primary system is composed of three primary coolant pumps, six decay heat removals, six intermediate heat exchanger. The secondary system has six intermediate loops comprised of one intermediate heat exchanger and six sodium/water heat exchangers. Then, the tertiary loops thirty-six separate loops. For simulation the coastdown transient, they used well known TRACE code and observed some discrepancy in their results and lack of inclusion of pump mechanical friction loss responsible for it.

The coastdown transient model in research and power reactors was extended to simulate the other core parameters such as the flow rate, temperature, pressure and departure from nucleate boiling ratio [23]. It is worth mentioning that no complicated equation like RELAP is solved and whole scenario is based upon the pressure drop relation and pump moment balance equation. It can be seen that with increasing time these parameters getting departure in comparison with experimental data. The reason might be the ignoring the pump mechanical friction loss like other models discussed earlier in the article. Then the coastdown transient was analyzed for a floating nuclear power plant by solving the complete pump speed modeling equation. A first-degree polynomial was fitted to measure the pump's internal part friction factor. The results for different energy ratios were presented for the flow rate and the speed of the pump [24]. It was suggested to include the a model for frictional torque in governing equations for minimizing discrepancies [12, 14, 16, 17, 19, 22, 23]. A quasi-steady state approach was introduced to simulate other parameters such pressure, temperature and DNBR and an excellent agreement with the experimental data was obtained. To qualify safety features of Super-Fast Reactors (Super FR), accident and transient's analyses were performed using validated computer codes SPRAT and SCRELA. For the partial loss of feed water, one of the reactor-coolant pumps trips with a coastdown time of 5 s, which leads to a decrease in the feed water flow rate by 50%. Signal of low flow rate (90%) is detected at the beginning, which leads to scram actuation. For the total loss of feed water flow accident, the scram signal is released at 0.5 s and the power rapidly decreases to the decay heat level in about 1 s. Scram actuation and/or auxiliary feed system actuation have been found effective to mitigate all abnormal transients in the Super FR [25, 27].

Additionally, a pump model for Sodium cooled fast reactor was solved using the TRACE code and argued to incorporate the friction factor in the model equations as well[22]. So, a novel method for the coastdown event by exploiting the pressure drop relation and pump movement balance equations was articulated [28]. In this approach the primary coolant flow and the pump speed governing equations were derived with the inclusion of frictional torque caused by the pump revolving part. Remember that the complete pump modeling equation was solved has the following form:

$$I\frac{d\omega}{dt} = \mathrm{M}_{em}(\omega) - \mathrm{M}_{h}(\omega) - \mathrm{M}_{f}(\omega)$$

where Mem(ω) stands for the electromagnetic torque and, $\mathrm{M}_h(\omega)$ represents the hydraulic torque and $\mathrm{M}_f(\omega)$ is for the friction torque. Moreover, $I\frac{d\omega}{dt}$represents the inertial torque of a rotor, I stands for the moment of inertia of the reactor coolant pump, ω is the angular velocity of the motor of the coolant pump respectively. It is worth mentioning that in the most of the previous studied the frictional torque of the pump is ignored which has shown a major flaw in the results of the previous studies, usually, in the latter half of the accident. But, in the study [28], the complete pump modeling equation is solved and the friction factor in the frictional torque is estimated with the second order polynomial and is derived using the concept of pump characteristics [29]. Resultantly, the key parameter, the effective energy ratio which interlinks the kinetic energies of the pump and of the primary coolant flow has become variable as the accident proceeds. The resulting method is applied to the Pakistan research reactor (PARR-1) and the Chashma-2 nuclear power plant (CHASNUPP-2) and plausible matching are found between experimental and theoretical results over longer time.

It is clear from [28] that the simulated results for the two pumps are showing off trend; the reason is that one pump has stopped during the course of the transients. Surprisingly, the results are for better that [12, 14, 16,17] because the visible off trend can be felt in latter half of the accident. Recall that the detail model is used for the present simulation.Similarly, the experimental study [30] to explore characteristics of coastdown transient following the loss of off-site power revealed that inertia of coolant pump should be high so that the speed of the pump and flow rate degrades slowly. This conclusion has been drawn by changing the inertia of the pump and seeing its effects of pump rotatory speed and the flow rate. In another study, a mathematical model of idling speed and flow characteristic curve of reactor coolant pump under the power failure condition was suggested and orthogonal optimization schemes was used with hydraulic modeling database of different vane structure parameters. The predictions from multiple linear regressions were compared with the experimental data [31].

Conclusion

Summarizing, the different simulated models can be divided into two types- the detail models and the short models. The detail models take into account all the pump parameters like pump speed modeling equation the pump characteristic curves and has the ability to predict results closer to the experimental data in comparison with short models. The close observation of the results of the different models has revealed that the Inertia of the Flywheel must be enough that the coolant pumps have ability to flood the core for the enough period until the Generator System becomes in the full operation. Also, it has been observed that the most models ignore the friction torque and consequently that results are comparable to experimental data for the longer time. Additionally, some discrepancy in the results can be observed at the initial stage of the accident [12, 14, 16, 17, 19, 22, 23]. So the inclusion of the frictional torque and homologous curves are must for predicting the correctness of experimental data, [28]. Also, some of the coastdown transient models has the capacity to predict the core decay heat to the coolant [19, 23]. These models take into account the simple equations and utilize the different parameters like the pump half time, the core time constant, the coolant half time, surprisingly, simulated the core outlet temperature comparable to the experimental data.

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