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Evaluation of Anticipated Transient Without Scram initiated by Total Loss of Reactor Coolant Flow

Saud Abalkhail[a*], Yongjae Lee[b], Kyoo Hwan Bae[b]
[a]*King Abdullah City for Atomic and Renewable Energy (K·A·CARE), Riyadh 11451, P.O. Box 2022*
[b]*Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Republic of Korea*
s.abalkhail@energy.gov.sa

1. Introduction

SMART (System integrated Modular Advanced ReacTor) is a fission nuclear reactor that abides by pressurized water reactor (PWR) design rules since it is a small sized PWR. The inherently safe features of passive safety systems are implemented in the design of SMART. In addition to the implementation of passive safety systems, SMART's core is designed to have what is known as a negative feedback loop, which takes the core power back to a safe mode.

This paper sheds light on the Anticipated Transient Without Scram (ATWS) initiated by the total loss of reactor coolant flow (TLOF) event. It is listed under Beyond Design Basis Accidents (BDBA), which allows for the best-estimate analysis methods. ATWS can be caused consequential to the various initiating events. The analyses of ATWS events are performed for the representative pressure increasing events in SMART such as the loss of normal feedwater flow, total loss of reactor coolant flow, and uncontrolled CRA withdrawal. These initiating events are design basis events (DBEs) normally but the fact that they are followed by the inability to SCRAM makes them BDBAs by definition of ATWS.

10 CFR 50.62 dictates that all PWRs must have a diverse protection system (DPS) that shuts down the reactor. Scram is performed by interrupting the power supplied to the control rod drive mechanism (CRDM) resulting in the drop of CRAs into the core, ultimately shutting it down. 10 CFR 50.62 also states the criteria for the fuel centerline temperature and the system pressure, where the centerline temperature must not exceed 2200 °F (1204.4 °C) and the system pressure must not exceed 110% of the design pressure. ATWS results in the over pressurizing of the RCS due to the imbalance of heat generation in the core and heat removal by the secondary side of steam generator (SG). In consequence, the pressurizer safety valve (PSV) should reach its setpoint and open, leading to system depressurization. However in SMART, this setpoint is not reached. This, in fact, should not be the case if the reactor protection system (RPS) is functioning, where the reactor trip setpoint would have actuated the shutdown control rods and the reactor core would be tripped. But this analysis considers the RPS to be dysfunctional and the diverse protection system (DPS) performs the reactor trip function. As the coolant temperature increases, the core power decreases due to the negative moderator temperature coefficients. The DPS also generates a passive residual heat removal actuation signal (PRHRAS) when the predetermined setpoint is reached. The PRHRAS generated by the DPS opens the PRHRS outlet isolation valves and closes the main steam isolation valves and the feedwater isolation valves. Natural circulation through the PRHRS cools down the reactor assisting it to reach safe shutdown conditions.

2. ATWS safety and mitigation system

In the event of any anticipated operational occurrences (AOOs), reactor trip occurs by the RPS and the residual heat of RCS is removed by the actuation of PRHRS. However, ATWS assumes that the scram function has been lost due to a dysfunctional RPS. Therefore, 10 CFR 50.62 has implemented a rule that the DPS should be installed in addition to the existing reactor trip system to provide diverse reactor scram, automatic initiation of turbine trip and auxiliary (or emergency) feedwater system. The DPS should be installed in addition to the existing reactor trip system to provide diverse reactor scram, automatic initiation of turbine trip and auxiliary (or emergency) feedwater system. This is why SMART employs the DPS to carry out these functions independently of the RPS.

Like the RPS, the DPS opens the output contactor in the motor-generator (MG) set of CRDM to drop the control rods into the core as well as generate the PRHRAS when a predetermined setpoint has been reached. The DPS uses a 2/2 logic circuit to execute this function. It is also worth mentioning that the DPS setpoint is the high pressurizer pressure, which is still lower than the PSV opening setpoint. In other words, the DPS will operate and reduce the core power and the reactor coolant temperature and pressure before the PSV comes into play to depressurize the system. This ensures that the reactor is kept safe from losing coolant through the opening of the PSV and possibly uncovering the core.

3. Analysis methodology

3.1 Analysis software:

TASS/SMR-S is a system thermal hydraulics. developed by KAERI as is being used for all safety and performance analysis purposes in SMART. TASS/SMR-S models the plant using nodes and paths, and calculates the thermal-hydraulic responses of plant, fuel rod heat flux and temperature. As for neutronics analyses, TASS/SMR-S utilizes the point kinetics model, which assumes the reactor core to be a single point and analyzes the reactor kinetics accordingly. This is also used to simulate the core power and the heat transferred to the coolant.

3.2 Analysis of the initial conditions of ATWS-TLOF:

Best estimate analysis is used for the analysis of ATWS because it is considered to be a BDBA. In this type of analysis, nominal values of thermal hydraulic parameters and kinetics parameters are used. We are mostly concerned about the view point of certain parameters:

- Core power
- Minimum departure from nucleate boiling ratio (MDNBR)
- RCS pressure

3.3 Choosing the most limiting case:

In ATWS, the major concerned safety parameters are the RCS pressure and MDNBR. For this reason, the sensitivity analysis to select the worst possible combination of MDNBR and system pressure to analyze further with the other parameters is performed. With that, a comparison between the top and bottom axial offsets (Figures 1 and 2) was conducted to see which case had exhibited the most limiting cases of system pressure and MDNBR. The axial offset is the fraction that indicates whether core power is higher towards the top of the core or the bottom. The case with the higher pressure and the lower MDNBR were henceforth chosen for the analysis.

From Figures 1 and 2, it is clear that the case of the top skewed axial offset was chosen as the focal point of the analysis since it displayed the more limiting case of pressure and MDNBR between the two cases.

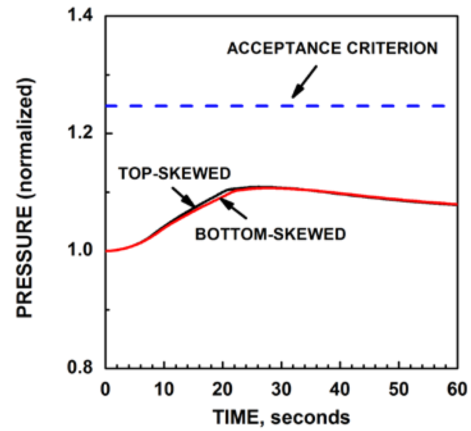

Fig. 1. Comparison of the normalized pressurizer pressure between the top and bottom skewed axial power

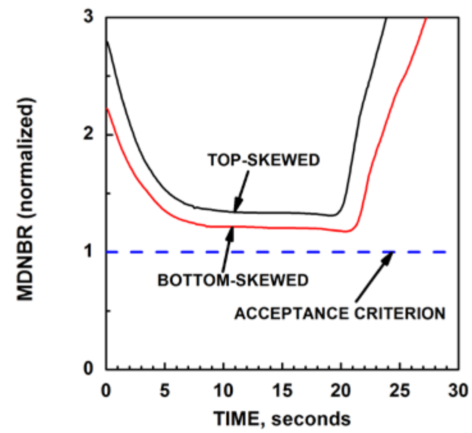

Fig. 2. Comparison between the MDNBR of the top and bottom skewed axial powers.

4. Analysis results and discussion

4.1 Reactor kinetics:

It is expected that the reactivity by the moderator temperature coefficient initially drops due to the density of the coolant being lowered. Also, the reactivity by the Doppler coefficient increases as the fuel temperature decreases due to the decrease in the core power caused by the negative MTC. This is seen in figure 3. It is observed that the DPS exhibits a major role in decreasing the reactivity insertion since the safety system causes a drop in the control rod into the core initiating scram.

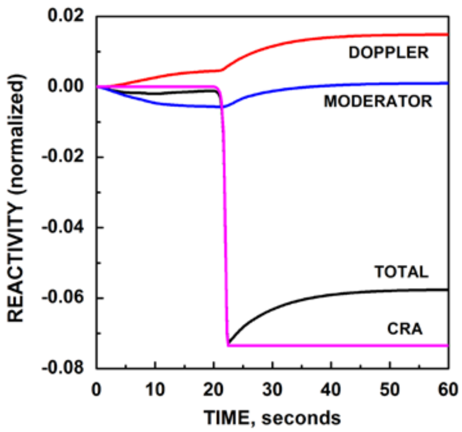

Fig. 3. Reactivity vs. Time

4.2 Core power:

When the ATWS-TLOF accident ensues, LOOP is assumed to be the cause of TLOF, initially. This entails that all RCPs stop working thus decreasing the RCS flow. This will cause an increase in RCS coolant temperature and pressure. In ATWS, the RPS is assumed to have failed so the DPS takes its place to trip the reactor core. For the DPS to actuate, the setpoint has to be reached. And the only setpoint in the DPS is the high pressurizer pressure. From there, the DPS is actuated and the reactor is tripped. It is seen in Figure 4 that the core power decreases rapidly upon the reactor trip.

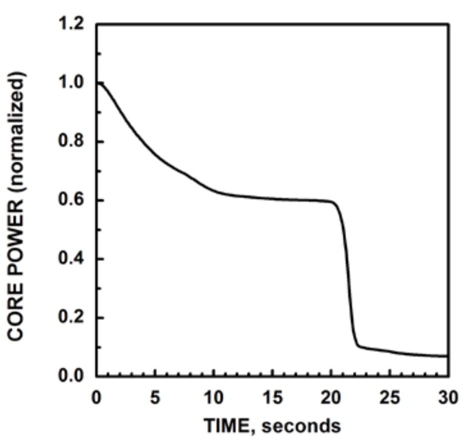

Fig. 4. Core power vs. time

4.3 MDNBR:

Related to core power, the MDNBR and the core inlet temperature will be altered in an event such as ATWS-TLOF. It is expected that the MDNBR decreases as the core heat flux increases due to the decrease in core flow rate. It would be fair to expect that the initial response of the core temperature to rise since the moderator temperature coefficient initially drops due to the density of the coolant being lowered as was observed. For this reason, the nature of the SMART reactor having a negative feedback loop entails that the Doppler coefficient, or FTC, drops as the MTC increases, and vice versa, decreasing the core temperature along the way. This will also decrease the MDNBR initially, as was mentioned previously but increase the value of MDNBR once more afterwards. This behavior is observed in Figures 5.

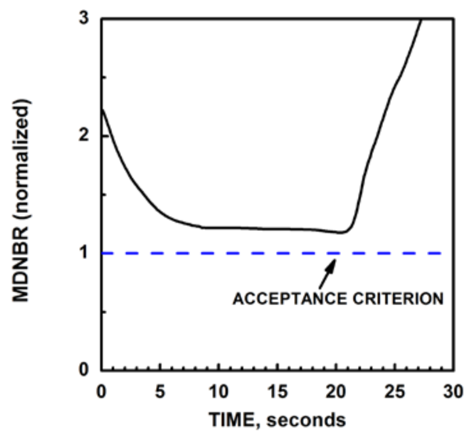

Fig. 5. MDNBR vs. time

4.4 RCS pressure:

Due to the stoppage of the RCPs in an accident such as ATWS-TLOF, the RCS and the PZR pressures would increase up to the reactor trip setpoint. From there, both the RCS and PZR pressures would have to drop due to the decrease in core power, causing the decrease in reactor coolant temperature. The RCS and PZR pressure should follow the same behavior at all times because they are connected to one another in the same vessel. It should be noted that the behavior should be the same but the actual value of the pressures might not be identical. The behavior is exhibited as was predicted in Figure1. It is safe to say that the acceptance criterion was met since the criterion requires that the pressure never to exceed 110% of the design pressure. The acceptance criterion line illustrated on Figure 6 is below the value that the criterion requires.

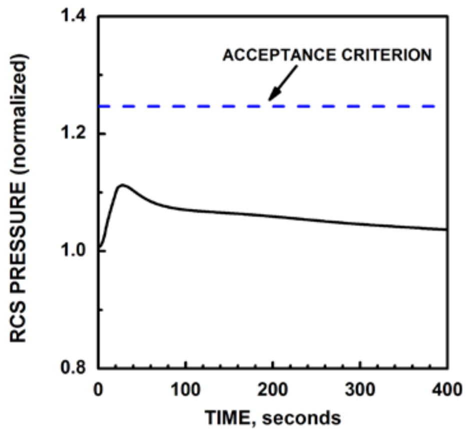

Fig. 6. PZR pressure vs. Time

4.5 PRHRS and long-term cooling:

When the PRHRAS is generated by the DPS, the PRHRS outlet isolation valves are opened and the main steam isolation valves and the feedwater isolation valves are closed. Thus, the PRHRS is connected to the SG directly. Natural circulation occurs due to the fact that the PRHRS is placed at a higher elevation than the SG. This allows for the lower density steam to flow from the SG into the heat exchanger in the PRHRS causing it to condense into liquid. The water would then drop and flow back down into the cold side of the SG due to gravity, where it would boil back into steam and rise up due to steam having lower density than liquid. This completes the cycle from the SG to the PRHRS and back while ensuring passive cooling for as long as it is required. Figure 7 shows the transient behavior of the PRHRS flow rate, which indicates that the natural circulation of fluid in the PRHRS is incurring.

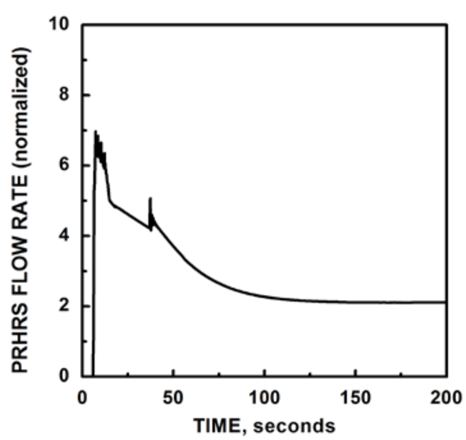

Fig. 7. PRHRS mass flow rate vs. Time

5. Conclusion

ATWS-TLOF is a transient that takes into consideration an AOO event accompanied by a dysfunctional RPS. The fail of scram function by RPS means that there should be a diverse protection system to drop the scram control rods as well as to actuate the PRHRS for the RCS cooldown. The DPS scrams the reactor in the case of losing the RPS functionality. To ensure that the DPS performs its role while holding the acceptance criteria at high priority, a number of analyses were performed to analyze the DPS in the most limiting conditions. As seen in the analysis results, the DPS performed its safety function well, maintaining safety margins and keeping the reactor in a safe condition throughout the transient.

REFERENCES

[1] 10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) events for light-water-cooled nuclear power plants.

ACKNOWLEDGEMENTS

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