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A Comparative Study between a SGTR and a SG Module Pipe Break Accident of the Integral Type Reactor

H.K. Kim[a], Y.J. Chung[a], G. H. Lee[a], H.C. Kim[a], S. Q. Zee[a]
[a]*Korea Atomic Energy Research Institute, P.O. Box 105, Yuseong, Daejeon, 305-600, Mail: hkkim@kaeri.re.kr*

1. Introduction

The integral type reactor, the small sized integral type pressurized water reactor with the rated thermal power of 65.5 MWt is one of the advanced types of small and medium reactors [1].

This study consists of a Steam Generator Tube Rupture (SGTR) and Steam Generator Module Pipe Break (SGMPB) analyses for the integral type reactor. The major concern of a SGTR is not the minimum Critical Heat Flux Ratio (CHFR) but the maximum integrated break flow from the primary to secondary side of the SG. The analysis of a SGMPB is performed to identify the minimum CHFR because the maximum integrated break flow is less than that of the SGTR accident for any case.

2. Analysis Methods

The definition of a SGTR means one helical tube rupture of a SG in the reactor vessel. The initiating event is one helical tube rupture in a SG. The fluid in the secondary system is mixed with that of the primary system including radioactivity. The mixed fluid is continuously sent to the turbine until the main steam isolation valve is closed. This radioactivity can be released to the environment by the air ejector in a condenser after sending it to the condenser. The air ejector is used for the release of the noncondensible gas to the atmosphere. Actually the reactor trip signal will be actuated soon by the radioactivity detectors on the steam lines. This signal indicates a high level leakage of radioactivity from the secondary system. For a conservative result, a reactor trip signal by an operator is used after not taking into account a signal of a high level radioactivity at the secondary system in the beginning. The system is tripped after 30 minutes from the initiating event by the operator trip signal. After signaling a reactor trip the SGs are isolated by the feedwater and the main steam isolation valves. And the SGs are connected to the Passive Residual Heat Removal System (PRHRS). The PRHRS removes the decay heat by a natural circulation.

The parameters of concern for a SGTR are the integrated break flow from the primary to the secondary system and the fuel integrity. In view of the break flow, a smaller break area creates a larger integrated amount through the main steam lines. Because the small break area brings about a delay of the reactor trip time.

The definition of a SGMPB is one module pipe break of a SG cassette in the reactor vessel. The penetration parts of the reactor vessel are the subsection pipes of the feedwater lines and the main steam lines. The module pipes in a reactor vessel act as a protective barrier for any radioactivity propagation from the primary to the secondary system.

One feedwater inlet module pipe break at a SG is the initiating event. The fluid in the secondary system is mixed with that of the primary system which includes radioactivity. For a conservation of the result, a reactor trip signal by a low Pressurizer (PZR) pressure is used after not taking into account the signal of a high level radioactivity in the secondary system in the beginning. The pressure of the Reactor Coolant System (RCS) is continuously decreased according to the leakage of the coolant to the secondary system. The system is tripped by a low PZR pressure signal. After signaling a reactor trip, the SGs are isolated by the feedwater and the main steam isolation valves. And the SGs are connected to the PRHRS.

The parameters of concern for a SGMPB are the fuel integrity and the system pressure. From the integrated break flow view point the maximum leakage of the SGMPB is similar or less severe when compared to the result of a SGTR. One stuck rod having the largest reactivity and Loss Of Offsite Power (LOOP) are assumed for an analysis of the conservative SGMPB analysis. As a result of a LOOP the Main Coolant Pump (MCP) does not supply coolant to the RCS after the reactor trip.

The analyses of a SGTR and a SGMPB are performed using the TASS/SMR code [2]. The ABAQUS code is used for the calculation of the fuel temperature [3]. The SGTR and SGMPB accidents are classified as the limiting condition accidents in the Safety Related Design Basis Events (SRDBE) for the integral type reactor.

3. Analysis Results

The sensitivity study of a SGTR for a break size is undertaken to find the maximum integrated break flow. The various initial conditions with the double ended break are considered at first. The longest time case until a reactor trip gives the maximum integrated leakage amount. As a result, the determined initial conditions are a high core power, high PZR pressure, high primary flow and a high coolant temperature. In the double ended break case, the integrated break flow is rather small because the reactor trip signal on the low primary system pressure occurs early. The smaller break size creates a greater integrated break flow due to the delay of the reactor trip signal. The least negative coefficient for the moderator temperature and the most negative coefficient for the Doppler reactivity are used.

By the break the core power decreases continuously until the reactor trip time. The maximum integrated break flow is found at 14.2% of the full break size. The integrated steam flow in the broken section is 11,740 kg in Fig.1. The primary system pressure is decreased continuously by a leakage of the coolant to the secondary system during the transient. The initial pressure, 15.51 Mpa, is the maximum primary system pressure and the minimum CHFR is 1.90. The hottest temperature of the fuel rod is 542 $^{\mathbf{o}}$**C**.

The sensitivity study of a SGMPB for the various initial conditions with the double ended break is performed from the fuel integrity viewpoint. As a result, the determined initial conditions are a low primary flow, high core power, high PZR pressure and a high coolant temperature. The least negative Doppler reactivity coefficient and the most negative moderator temperature coefficient create severer results. The reactor trip signal set by the low PZR pressure is generated at 65 seconds after the initiating event occurs.

According to an over-cooling due to a mismatching of the heat generation in the core and the heat removal in the SG, the coolant temperature at the SG inlet decreases from the beginning of the transient. Thus, the core power is increased by the negative reactivity characteristics of the coolant. The core power has a maximum value at 26 seconds of 119% of the nominal power.

The RCS pressure decreases continuously by a leakage of the coolant to the secondary system during the transient. The initial pressure, 15.52 MPa, is the maximum RCS pressure. The RCS pressure is stabilized by the PRHRS after the reactor trip. The pressure at the broken part of a SG is increased by the incoming flow from the RCS at an initial time period. By the initiation of the PRHRS, the secondary pressure maintains a balance with the pressure of the RCS.

The CHFR decreases by the feedback effect of the reactivity at an initial time period. The minimum CHFR is found to be 1.73 before the reactor trip at 24.35 seconds in Fig 2. The hottest temperature of the fuel rod is 481 ^{o}C.

4. Conclusion

The comparative study for a conservative calculation of the SGTR and SGMPB accidents in an integral type reactor are performed using the TASS/SMR code. The minimum CHFRs are maintained at over 1.3, which is one of the acceptance criteria, during the transient. The hottest temperatures of the fuel rod are far less than the temperature of the design limit throughout the transients [4].

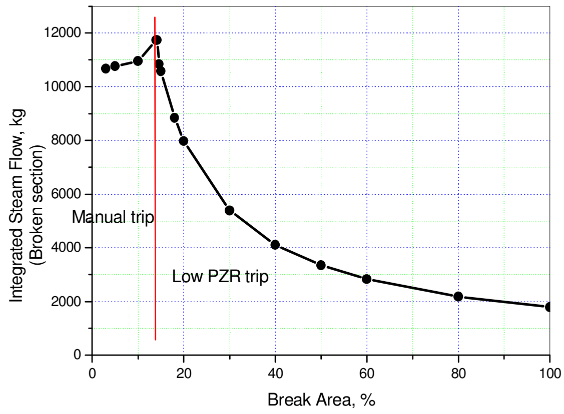

Figure 1 Integrated Steam Flow of SGTR

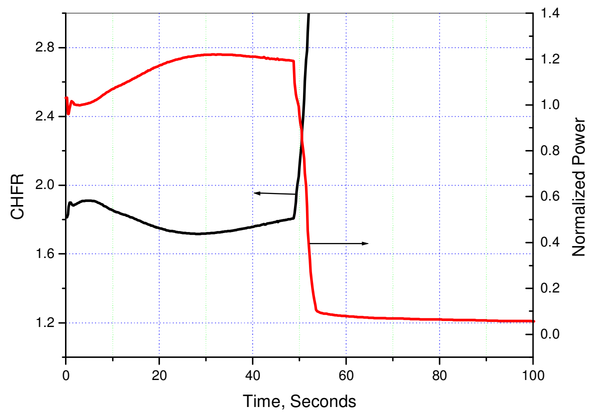

Figure 2 CHFR and Power of SGMPB

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REFERENCES

[1] M.H. Chang, et al., “Pre-study report for an integral reactor development project (SMART-P)”, KAERI/RR-2260/2002

[2] H.Y. Yoon, et al., “Thermal Hydraulic Model Description of TASS/SMR”, KAERI/TR-1835/2001

[3] D.H. Hwang, et al., “Development of CHF correlation systems for SMART-P fuel assembly”, KAERI/TR-2943/2005

[4] H.K. Kim, et al., “Methodology for the pipe break accident analysis in steam generator for the SMART-P”, KAERI/TR-2941/2005