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Methods and Effects of Safety Enhancement in Korean PSR

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1. INTRODUCTION

Periodic Safety Review (PSR) is a comprehensive study on a nuclear power plant safety, taking into account aspects such as operational history, ageing, safety analyses and advances in code & standards since the time of construction [1]. In Korea, PSRs have been performed for 20 units and have been effectively used to obtain an overall view of actual plant safety to determine reasonable and practical modifications that should be made in order to obtain a higher level of safety approaching that of modern plants.

Among many safety enhancements achieved from Korean PSRs, new safety analyses are the important methods to confirm plant safety by increasing safety margin for specific safety issues. Methods and effects of safety enhancements applied in Korean PSRs are reviewed in this paper in light of new safety analyses to obtain additional safety margins.

2. SAFETY ISSUES AND METHODS OF SAFETY ENHANCEMENT

Regulatory positions, the methods and the effects of each safety enhancement are presented in Table 1 and are described in detail below.

<u>Transient Events</u>

Major safety issues related with plant transient events raised during PSRs are upgrade of the core source term, atmospheric dispersion of radioactivity, and the test tolerance of the safety valves.

For the first issue, an offside dose assessment is required according to the fuel cladding gap source term of USNRC Regulatory Guide 1.25 which has been imposed on new plants in Korea. For the atmospheric dispersion, USNRC Regulatory Guide 1.145 revision 1 has been newly applied to the offsite dose assessment after accidents. The new guide requests more stringent modeling of the atmospheric dispersion factor and this has increased the values.

Nevertheless, for the two offsite dose issues, the 10CFR100.11 acceptance criteria have been satisfied by applying new dose conversion factors from ICRP-2 to ICRP-30 and/or decreasing the primary to secondary coolant release rate designated in the technical specification.

For the safety valve test tolerance issue, the Korean regulatory body requested an increase of the tolerance in accordance with USNRC generic safety issues 165[2]. For this, more realistic safety analyses for the overpressure transients such as loss of load and feedwater line break were performed to obtain a greater safety margin with respect to the peak system pressure ; this enabled 2~4% test tolerances, a substantial improvement over the original 1%.

<u>Loss of Coolant Accident</u>

For a LOCA, two major issues have been raised through the PSR. One is the post-LOCA long term cooling performance with respect to boron precipitation and debris deposition on the core internals. The other is the effect of optimization of safety system operation, e.g., spray termination on the containment integrity in terms of decreasing the transport of fragments generated by the break force.

New analyses using RELAP5 [3] were performed to evaluate the effect of boron precipitation and the chemical deposition at the core entrance and the fuel sheath. It was assumed that the core entrance is 95% blocked by foreign materials and the fuel sheath heat transfer to the coolant is reduced by 10%. The results show that the long term cooling of the core is maintained with these extreme assumptions. Also, tests have been performed for boiling water with simulated chemical deposits on the specimens of the rector internals, revealing that deposits have a negligible effect on the boron precipitation limit.

To evaluate the effect of optimization of safety system operation, LOCA analyses have been performed with the assumption of spray termination after recirculation. It was shown that the containment pressure and temperatures are maintained below the curves assumed for equipment qualification with sufficient safety margin.

<u>Total Loss of Feedwater</u>

OPR1000s have safety depressurization systems (SDS) installed on a pressurizer to mitigate the event of total loss of feedwater (TLOFW). The acceptance criteria applied in the design of the SDS, as stated in the Safety Review Guide (SRG) Appendix 19.1 'Rapid Depressurization Capability' [4] of the Korean regulation, is that the core should not be uncovered by bleed and feed (B&F) operation using only safety grade systems assuming a single failure.

Although not stated in the SRG App. 19.1, the quantitative criterion of core coverage applied in the KSNPs is that the core mixture level should be maintained at least two feet above the core top plate by B&F operation under a TLOFW event.

There are no SDS in the Westinghouse type and Framatome type reactors in Korea. Thus PORV operation is assumed for the B&F operation and the analyses were carried out using the best-estimate code RELAP5. The results show that the core mixture level is maintained two feet above the core top plate if the B&F operation is started no later than 15 minutes after initiation of an event with two of three PORVs and two high pressure safety injection pumps.

Anticipated Transient without Scram

The USNRC rule of 10CFR50.62 [4] requires that a pressurized water reactor must have equipment from the sensor output to final actuation device, reliable and independent from the existing reactor trip system, to automatically initiate the auxiliary feedwater system and a turbine trip under ATWS. According to the Korean regulation, which follows the 10CFR50.62, AMSAC(ATWS Mitigation Signal Actuation Circuitry) has been newly installed in old Korean plants and new backup analyses have been performed and they showed that period of unfavorable MTC is less than 5%.

Table 1. Safety enhancements in Korean PSRs

	Safety Issues and Regulatory Positions during PSRs	Method of Safety Enhancements	Effects of Safety Enhancements
Transients Events	● Upgrade of core fission product source term ● Apply new requirement on atmospheric dispersion factors ● Increase the safety valve test tolerances	● New analyses according to new requirements that have been imposed on new plants in Korea ● Change dose conversion factors and/or decrease primary-to-secondary release ● More realistic safety analyses of the overpressure transients	● Change dose conversion factors and/or decrease primary-to-secondary release increased safety margin ● 2~4% tolerances were obtained
Loss of coolant accidents	● Evaluate the post-LOCA long term cooling with respect to boron precipitation and foreign material deposition on the reactor internals ● Evaluate the effect of spray termination on the containment integrity	● Analyses assuming crust deposition on the fuel sheath and 95% blockage of core entrance ● New test on the effect of chemical deposition on the boron concentration limit ● New analyses of containment pressure and temperature after spray termination at the time of recirculation	● Negligible effect of the chemical deposition and boron precipitation on the cooling ● Negligible change of boric acid precipitation limit by chemical deposits ● Containment pressure and temperatures are maintained below the curves of equipment qualification
Total loss of feed water	● Confirm the core mixture level is maintained two feet above the core top after bleed and feed operation assuming only safety grade systems and single failure	● New analyses using best-estimate codes	● Bleed and feed operation when initiated within 15 minutes with two of three PORVs and two safety injection pumps meet the requirement
Anticipated transient without scram	● Need reliable and independent signal to automatically initiate the emergency feedwater system and a turbine trip ● Analyze the unfavorable MTC	● ATWS mitigation systems have been installed ● New analyses using best-estimate transient analysis computer codes	● New analyses showed that unfavorable MTC is less than 5% of one fuel cycle

3. CONCLUSIONS

Periodic safety review is an effective way to determine reasonable and practical modifications that should be made in order to obtain higher level of safety. To obtain enhanced safety when new requirements are applied, new safety analyses are needed to obtain an additional safety margin. This paper reviewed the experience of applying new requirements during PSRs as well as the methods and effects of safety enhancement by using new safety analyses.

REFERENCES

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[4] KINS, “App. 19.1 Rapid Depressurization Capability,” Safety Review Guide (SRG), Korea Institute of Nuclear Safety (KINS), 1999.