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Model and Estimation of ATWS Frequency for the APR-1400 Reactor

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1. Introduction

An anticipated transient without scram (ATWS) is an anticipated operational occurrence (AOO) that results in a rapid pressure rise of the primary side by no reactor trip. The magnitude and timing of the reactor coolant system (RCS) pressure rise depends on the moderator temperature coefficient (MTC), the pressure relief capacity and the energy removal capacity of the secondary side in the pressurized water reactor (PWR). It is dealt with an important safety issue in the point that the primary pressure over ASME stress C level (3,200psig) can lead to core damage consequently.

This paper focuses the modeling issues and estimation of an ATWS frequency for the APR-1400 reactor (e.g., Shin Kori 3&4), where a digital reactor protection system (DRPS) is installed [1].

2. Methods and Results

2.1 ATWS Frequency Model

To evaluate an ATWS frequency, the development of fault tree (FT) is required each reactor trip parameter for the APR-1400 DRPS as follows.[1]

- P1: Variable Over-Power Trip (VOPT)
- P2: High Logarithmic Power (Hi LOG PWR)
- P3: High Local Power Density (Hi LPD)
- P4: Low Departure from Nucleate Boiling Ratio (Lo DNBR)
- P5: High Pressurizer Pressure (Hi PZR PR)
- P6: Low Pressurizer Pressure (Lo PZR PR)
- P7&P8: Low Steam Generator(SG) #1&#2 Pressure (Lo SG PR)
- P9&P10: High SG #1&#2 Level (Hi SG LVL)
- P11&P12: Low SG #1&#2 Level (Lo SG LVL)
- P13: High Containment Pressure (Hi CTMT PR)
- P14&P15: Low SG #1&#2 RCS Flow (Lo RCS FW)

Except for the manual trip, the DRPS has 15 types of automatic trip parameters. For two digital signals of them (LPD and DNBR), system FT for core protective calculator (CPC) and Control Element Assembly Calculator (CEAC) are required. FTs for the high pressurizer pressure (Hi PZR PR) and high containment pressure (Hi CTMT PR) signals of the diverse protection system (DPS) are also needed. Figure 1 illustrates the high-level FT logic for the ATWS frequency.

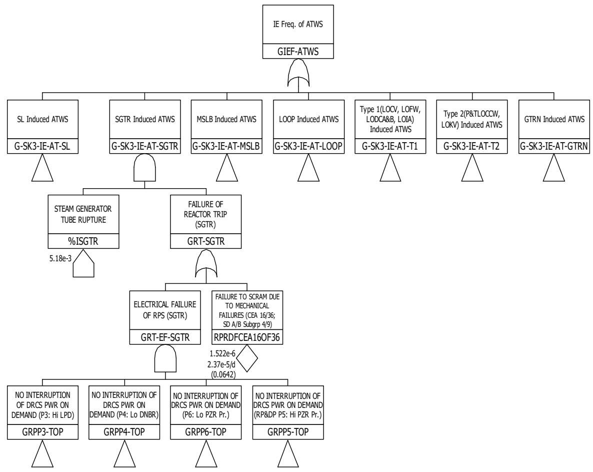

Figure 1. An illustration of the Top Logic of the FT for the ATWS Frequency

2.2 Major Issues and Ground-rules for ATWS Frequency Model

Major modelling issues and ground-rules for estimating the ATWS frequency are as follows:

① Identification of the reactor trip parameters followed each initiating event (IE)

- IEs not to need scram: large and medium loss of coolant accidents (LLOCA and MLOCA)
- IEs that scram is required without any reactor trip signal: loos of off-site power (LOOP) and station blackout (SBO)
- IEs leading to core damage directly regardless of scram in the PSA model: reactor vessel rupture (RVR) and interfacing system LOCA (ISLOCA)
- The remaining except IEs mentioned above: two digital signals (LPD and DNBR) are issued for all IEs. Also, the additional 3rd trip signals according to the IEs are considered as shown in Table 1. Note that they are assumed, based on the experimental results from the simulator of the OPR-1000 reactor[2].
- Finally, high pressurizer pressure (Hi PZR PR) signal are added for all IEs in the ATWS

frequency model. It is because the primary concern of ATWS is a rapid pressure rise of the primary side by no reactor trip. Of cause, backup trip signal by DPS is also considered together

Table 1. The 3rd Trip Parameters Assigned for IE groups

IE Group*	3rd Trip Signal
SLOCA	Hi CTMT PR.
SGTR	Lo PZR PR.
MSLB-IC, MSLB-OC	Lo SG LVL or Lo SG PR.
LOCV, LOFW, LOIA, LODCA, LODCB	Lo SG LVL.
PLOCCW, TLOCCW, LOKV	Lo RCS FW
GTRN	VOPT or Hi SG LVL

*) small LOCA (SLOCA), steam generator tube rupture (SGTR), main secondary line break-inside/outside containment (MSLB-IC/OC), loss of condenser vacuum (LOCV), loss of feed-water (LOFW), loss of instrument air (LOIA), loss of 125V DC bus A/B (LODCA/B), partial/total loss of component cooling water (P/TLOCCW), loss of 4.16KV bus (LOKV), general transients (GTRN)

② Success criterion for control element assembly (CEA) insertion

- The results of thermal-hydraulic (TH) analyses by MARS (Multi-dimensional Analysis of Reactor Safety) code for the OPR-1000 reactor [2]: the RCS peak pressure does not reach to ASME stress C level (approximately 220 bar) if CEAs with the reactivity worth of the 0.1% over insert into the core (Refer to Figure 2). Note that success criterion for the OPR-1000 reactor was determined as the insertion of any 3 groups (12 CEAs) among total 7 group for shutdown (28 CEAs for shutdown), considering the uncertainty of the results for TH analyses,

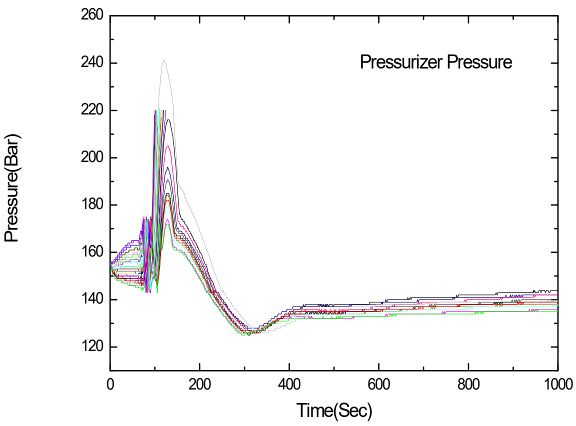

Figure 2. The Results of TH Analysis for Determining Success Criterion of CEA Insertion at OPR-1000 reactor

- There are total 9 groups for shutdown (36CEAs) in APR-1400 reactor. Finally, success criterion for the APR-1400 reactor is assumed as the insertion of any 4 groups (16 CEAs) among total 9 group for shutdown (36 CEAs for shutdown), considering the uncertainty

③ Operator error probabilities to fail reactor trip manually

- To obtain more realistic model for the post-accident operator error events, the manual reactor trip can be divided into two conditions; 1) no reactor trip due to mechanical failures of all TCBs, and 2) no automatic trip signal.
- For the first case above, the failure probability is estimated to be 0.032, based on the results of simulator experiments by 4 actual operating teams of an OPR-1000 reactor [2].
- For the second situation, it is assumed to be 0.5 considering manual reactor trip with some limited non-safety information available in MCR and the functional dependency factors, e.g., inconsistency between the safety and non-safety-related information. Note that it is more conservative value than 0.07 for OPR-1000 reactor[2].

④ The plant-specific FT models of the digital I&C system for APR-1400 reactor

- The plant-specific FT models for 15 reactor trip signals are developed, based on state-of-art digital I&C modeling technique and as designed/as-operated DRPS information of Shin-Kori 3&4.
- CPC and CEAC are included in FT models for P3 and P4. DPS trip signals for backup of P5 and P13 are considered appropriately.
- For more information of details, refer to the reference [3].

2.3 The Results and Findings

The results and the major findings are summarized as follows.

- The ATWS frequency for APR-1400 reactors are estimated as 1.4e-6/RY (point estimate).
- The two types of operator error probabilities related to manual reactor trip are very sensitive to the priorities of the minimal cutsets obtained from the ATWS frequency models.
- In 1983, Nuclear Regulatory Commission (NRC) required the additional facility installation to improve the plant capacity to prevent an ATWS

and mitigate its consequences, so-called the ATWS rule [4]. The position of NRC staff on the ATWS rule states that the core damage frequency (CDF) from an ATWS, so-called ATWS risk, has to be lower than 1.0e-5/ RY [5]. Note that the ATWS risk is simply defined as the multiplication of the ATWS frequency and unfavorable exposure time (UET). Even though we adopt a conservative assumption of the UET (about 33%) for the APR-1400 reactor, the ATWS risk is evaluated as 4.5e-7/RY, which copes with the intentional target of the ATWS rule (1.0e-5/RY) sufficiently.

3. Conclusions

The plant-specific ATWS frequency model for the APR-1400 reactor was developed using more realistic information and the state-of-art technology. Several plant-specific results analyzed for the OPR-1000 reactor are used to address the modeling and technical issues of ATWS frequency estimation for APR-1400 reactor. It needs further efforts for producing the plant-specific safety information for the APR-1400 reactor.

ACKNOWLEDGEMENTS

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