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ATWS Frequency for the Analog I&C System of the OPR-1000 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.

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 [1]. 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/reactor year (RY) [2]. Note that the ATWS risk is simply defined as the multiplication of the ATWS frequency and unfavorable exposure time (UET).

This paper focuses the estimation of an ATWS frequency for the OPR-1000 reactor with an analog reactor protection system (RPS). It is an important issue in risk-informed technical specification (RITS) of RPS [3].

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 OPR-1000 RPS as follows.

- Variable Over-Power Trip (VOPT)
- High Logarithmic Power (Hi LOG PWR)
- High Local Power Density (Hi LPD)
- Low Departure from Nucleate Boiling Ratio(Lo DNBR)
- Low Pressurizer Pressure (Lo PZR PR)
- High Pressurizer Pressure (Hi PZR PR)
- Low Steam Generator Pressure (Lo SG PR)
- Low Steam Generator Level (Lo SG LVL)
- High Steam Generator Level (Hi SG LVL)
- High Containment Pressure (Hi CTMT PR)
- Low Reactor Coolant Flow (Lo RCS FW)

Except for the manual trip, the PPS includes 11 types of automatic trip parameters for the RPS. 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 of the diverse protection system (DPS) are also needed. Figure 1 illustrates the high-level FT logic for the ATWS frequency.

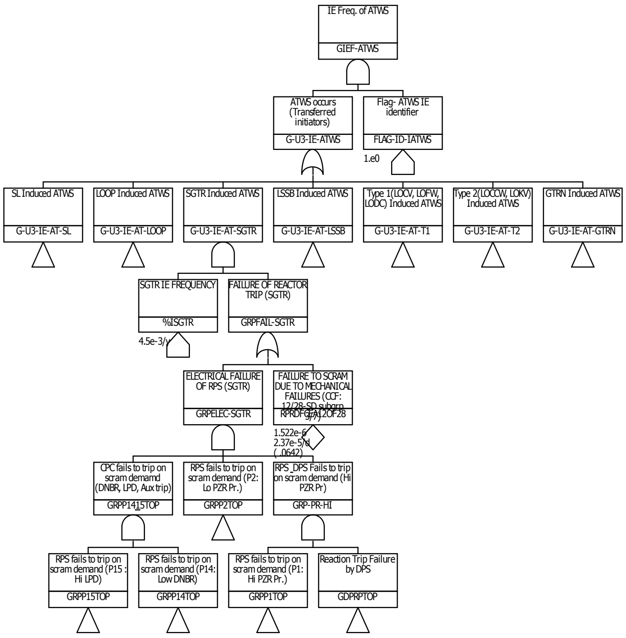

Figure 1. An illustration of the Top Logic of the FT for the ATWS Frequency

2.2 Major Improvements during the Model Development

Major improvement items in the model are as follows, compared with the previous model [4].

① Identification of the reactor trip parameters followed each initiating event (IE) from the results of simulator experiments.
  - IEs not to need scram : large and medium loss of coolant accidents (LLOCA and MLOCA)
  - IEs that need scram, but require no reactor trip signals: loos of off-site power (LOOP) and station blackout (SBO)

- IEs leading to core damage directly in the PSA model: reactor vessel rupture (RVR) and interfacing system LOCA (ISLOCA)
- The remaining IEs except mentioned above: two digital signals (LPD and DNBR) are issued for all IEs. Table 1 represents the additional 3rd trip signals according to the IEs.

Table 1. Plant Protection System Parameters

IE Group*	3rd Trip Signal
SLOCA	Hi CTMT PR.
SGTR	Lo PZR PR.
LSSB	Lo SG LVL or Lo SG PR.
LOCV, LOFW, LODC	Lo SG LVL.
LOCCW, LOKV	Lo RCS FW
GTRN	VOPT or Hi SG LVL

*) small LOCA (SLOCA), steam generator tube rupture (SGTR), large secondary side break (LSSB), loss of condenser vacuum (LOCV), loss of feedwater (LOFW), loss of DC bus (LODC), loss of component cooling water (LOCCW), loss of 4.16KV bus (LOKV), general transients (GTRN)

② Change of success criterion for control element assembly (CEA) insertion by thermal-hydraulic (TH) analyses.

- The results of TH analyses by MARS (Multi-dimensional Analysis of Reactor Safety) code: 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).
- Considering the uncertainty of the results for TH analyses, success criterion was determined as the insertion of any 3 groups (12 CEAs) among total 7 group for shutdown (28 CEAs for shutdown).

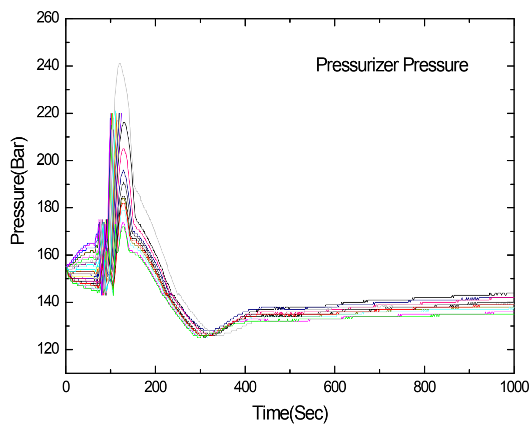

Figure 2. The Results of TH Analysis for Determining Success Criterion of CEA Insertion

③ Re-evaluation of operator error probabilities to fail reactor trip manually based on the results of simulator experiments

- To obtain more realistic model for the post-accident operator error events, the manual reactor trip was divided into two conditions; 1) no reactor trip due to mechanical failures of all TCBs, and 2) no automatic trip signal. The failure probability for the first case was estimated to be 0.032, based on the results of simulator experiments by 4 actual operating teams of an OPR-1000 plant. For the second situation, however, it was assumed to be 0.07 considering manual reactor trip by non-safety information in MCR and the functional dependency factor among the safety-related signals [5].

④ The use of the plant-specific operating experience analysis results

- The development of the reliability database (independent and common cause failures) for the safety-related I&C components, based on the operational data of the total 24.24 reactor years for the period of 2003 through 2007 at six OPR-1000 reactors ([6],[7]).
- The use of alpha factor method for modeling common cause failure (CCF)
- The estimation of maintenance and test unavailabilities based plant-specific operating information and practice.

2.3 The Results and Findings

The results and the major findings are summarized as follows.

- The point estimates of ATWS frequency for OPR-1000 reactors are ranged from 3.02e-6 ~ 3.14e-6/ RY.
- 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.
- Even though we adopt a conservative assumption of the UET (~ 0.33) for the OPR-1000 reactor, the ATWS risk defined by the NRC staff is evaluated as 1.0e06/RY that cope with the intentional target of the ATWS rule (1.0e-5/RY).

3. Conclusions

The plant-specific ATWS frequency model for the OPR-1000 reactor was developed using more realistic information and the state-of-art technology. The results of the work can be directly used to improve risk-informed surveillance test interval (RI-STI) of the KSNP safety-related I&C systems such as RPS.

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