Transactions of the Korean Nuclear Society Spring Meeting
Chuncheon, Korea, May 25-26 2006

Study of Risk-Informed Approach to Optimize Test Interval of RPS/ESFAS Systems for Kori Unit 2

Bag-Soon Chung, Sung-Yull Hong, Myung-Ki Kim, Hae-Cheol Oh, and Mi-ro Seo, Jeong-Heon Ha,
Korea Electric Power Research Institute, Korea Electric Power Corporation
103-16 Munji-Dong, Yusong-Gu, Daejeon, 305-380 Korea, bschung@kepri.re.kr

1. Introduction

Probabilistic safety assessment (PSA) has had a growing amount of use in the electric power industry. It has always been the position of most of the Nuclear Power Industry, including nuclear power plants in Korea, that PSA is to be used as a source of information for prudent decision-making. To relax test interval reactor protection system and engineered safety features actuation systems (RPS/ESFAS) for Kori Unit 2, a risk-informed approach has been studied since the project started August 2005. I would like to introduce the risk-informed approach to be used for relaxing the test interval of RPS/ESFAS systems of Kori Unit 2.

2. System Description

The reactor protection systems (RPS) circuit consist of analogue channels, combination logic units, and trip breakers. The engineered safety features actuation system circuits are composed of analogue channels, combination logics, and actuation relays. Fig. 1 shows the block diagram of RPS/ESFAS and the test points.

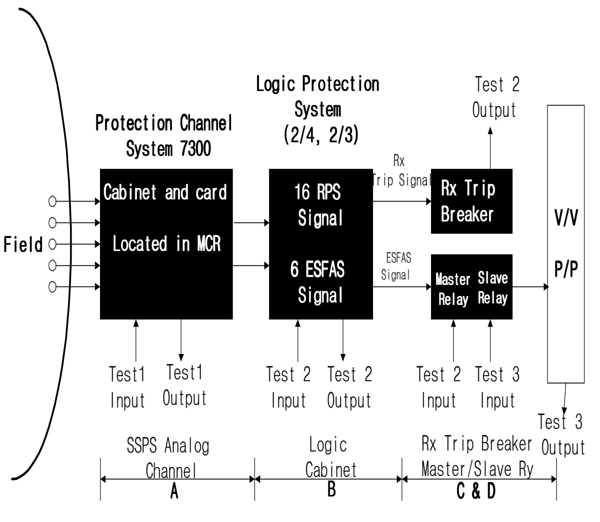

Fig. 1 Block Diagram of RPS/ESFAS for Kori Unit 2

3. System Fault Tree

To analyze the impact of increasing AOTs and STIs on system unavailability, a fault tree analysis of the individual functions for the RPS/ESFAS was performed. The five major contributors which effect on unavailability are 1) random failures 2) test 3) maintenance 4) Human Error 5) Common cause failure. The average unavailability of random failure during test interval T can be obtained by

$$\Pr(t) = \frac{1}{T}\int_0^T \Pr(t)dt \quad \approx \quad \frac{1}{2}\lambda\, t$$

where, $\lambda T < 0.1$. Therefore the unavailability is sensitive to the chosen test interval. The unavailability of a component due to test was calculated using the formula $P_t = \lambda_t T$, Where P_t is unavailability due to test and λ_t is the mean number of tests per hour and T is the mean duration of test. The unavailability of a component due to maintenance was calculated using the formula $P_m = \lambda_m T$, Where P_m is unavailability due to maintenance and λ_m is the mean number of tests per hour and T is the mean duration of maintenance. Human error such as miscalibration or misposition of a component was modeled in the fault tree. THERP (Technique for Human Error Rate Prediction) method will be applied to analyze human error probability. Common cause failure can be defined as simultaneous failure of like components with identical function requirements. For reactor trip breakers, master relay, logic cabinet, The Common cause failure probability will be calculated with equation of $P_{cc} = \beta \times P_r$, where β is the Beta factor, P_r is probability of random failure of component. For the probability of CCF of analogue channel, the MGL approach will be used

Fault trees were constructed to model the each signal of RPS/ESFAS to allow the calculation of the unavailability of individual trip functions. 22 RPS and 12 ESFAS signals were assigned in new model to the fault tree top gate. And it has modeled the detailed component failure event of sensors, nuclear instrument systems, reactor trip breakers, actuation relays. For sensitivity study, the current and the proposed STIs and AOTs will be evaluated in terms of core damage frequency and large early release frequency.

System fault trees were constructed for RPS and ESFAS actuation signals. And the level of detail in the fault included the analogue channel components such as sensor, signal comparator card, universal cards in SSPS, undervoltage card, reactor trip breaker, safety-guard driver card, and master/slave relay.

Test and maintenance outages and associated RPS/ESFAS configurations were modeled for the SSPS and channel outages. For channel outages, the channel is assumed to be placed into a bypass condition rather than a tripped mode. For SSPS train outages, the other train is available to respond to plant upset conditions.

4. Data Review and Analysis

For the failure database of each component, plant specific data of Kori Unit 2 will be obtained by using Bayesian update method for plant data and the Westinghouse data base, WCAP-10271.

RPS/ESFAS performance during the period 1996 through 2005 was assessed by reviewing trouble reports, reactor operator's reports, and channel calibration reports. The data review process involves at least two independent reviews of each report by knowledgeable engineers. Data analysis for the component failures involves several steps:

a. Demand count and exposure time estimation
b. Statistical analysis of data subgroups to identify differences
c. Components unavailability estimation
d. CCF events unavailability estimation

The components demand counts were estimated from plant scram histories, and testing intervals.

5. Risk Analysis

The risk analysis will be carried out to determine the impact of changes in AOTs, STIs on plant safety. It is necessary to assess the impact of the changes on plant safety to establish a measurable impact. The unavailability analysis provides the impact of the changes on signal availability, but it is not possible to draw conclusions since it can not point out how important the signals are to plant safety. The risk model is quantified with the KIRAP code to calculate core damage frequency. The base case will be initially quantified with the signal unavailability corresponding to current AOTs and STIs. These will be followed by quantifications with the signal unavailability for each case. In addition to that, unnecessary plant transients and challenges to the protection systems caused by test will be considered. An evaluation of core damage frequency/large early release frequency caused by forced outages that occurred from the commercial operation date of KORI Unit 2 will be performed.

6. Deterministic Considerations

In addition to risk information of plant such as system unavailability, core damage frequency and large early release frequency, change to the test intervals will be determined from consideration of degradation of system/equipment caused by testing and manufacturer's specification and recommendation, performance of equipment in similar plants or environment as required in ANSI/IEEE Std. 338-1987(IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems).

7. Conclusions

Risk-informed STI/AOT modification technology is established with the study. The results of the study are used to prepare the licensing submittals for proposing to extend the current requirements of STI/AOT for RPS/ESFAS of Kori Unit 2. This project will contribute to reducing the plant staff's burden to perform the test, and to prevent the adverse effects to safety caused by human error during the test.

REFERENCES

[1] Evaluation of Surveillance Frequencies and out of service times for the reactor protection instrumentation system, WCAP-10271, May 1989
[2] Probabilistic risk analysis of the RPS and ESFAS test times and completion times, WCAP-14333-P, May 1995
[3] Risk-informed Assessment of the RTS and ASFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times, WCAP-15376, October 2000
[4] RPS/ESFAS Tech. Spec. Optimization Study for Kori Unit 3,4 & Yonggwang Unit 1,2, September 1999
[5] IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems, ANSI/IEEE Std. 338-1987