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A Risk Informed Approach to Relax AOTs and STIs in Technical Specifications

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Abstract

A risk informed approach to relax AOTs and STIs of RPS/ESPAS in technical specifications of Kori units 3,4 was performed in this paper. With the proposed AOTs and STIs, system unavailabilities and core damage frequency were quantified using PSA model. The results show that the core damage frequency is slightly increased by extending AOTs and STIs but negligible. As considering the benefits such as reduction of plant transients and man power for test and maintenance, the relaxation of AOTs /STIs of RPS/ESFAS is justified.

Ⅰ. Introduction

With respect to the impact of current testing and maintenance requirements on operating plants and particularly that of reactor protection system(RPS) and engineered safeguard feature actuation system(ESFAS), probabilistic safety assessment(PSA) is currently recognized as a good approach for relaxing technical specifications(TS). TS are safety rules for nuclear power plants approved by regulatory authority. The surveillance test intervals(STIs) and allowable outage times(AOTs) specified in TS have been developed based on results of deterministic analysis and engineering judgement than on risk calculations.

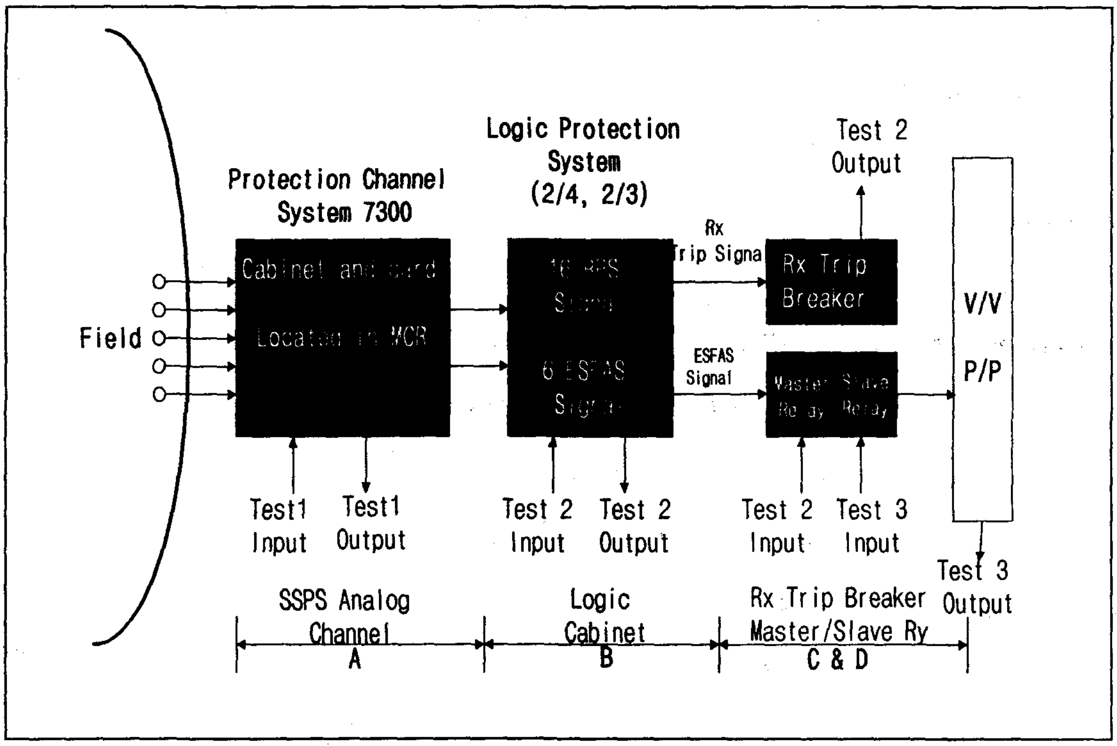
The instrument channels, interlocks of RPS and the automatic actuation logic and relays of ESFAS shall be demonstrated operable by the limiting conditions for operation and surveillance requirements in FSAR. But operating plants have experienced many inadvertent reactor trips during performance of testing, causing unnecessary transients and challenges to safety systems. Also significant time and effort on an operating staff must be devoted to performing, reviewing, documenting and tracking various surveillance activities.

To provide justification for relaxing AOTs and STIs, a risk informed approach was introduced in this paper. This paper is organized as follows: The design, test & maintenance features of RPS/ESFAS were briefly described in section II. In section III, the system unavailability was analyzed on changing in AOTs and STIs. The plants risk was evaluated with proposed STIs & AOTs by CDF in section IV and in addition, the qualitative aspects of system modifications and preventive maintenance activities to increase I&C sytem reliability were also described. Section V summarizes conclusions.

II. System Description

The reactor protection system circuit consists of analog channels, combination logic units, and trip breakers. ESFAS circuit is composed of analog channels, combination logic, and actuation relays. Fig.1 shows the block diagram of RPS/ESFAS and the test points.

Figure 1. RPS/ESFAS block diagram.



The analog channels provide signals to logic cabinets, which provide signals to reactor trip breakers and the actuation relays. The actuation relays consist of master and slave relays. The master relays are controlled by the logic cabinet and the slave relays are controlled by the master relays. The slave relays actuate the required equipment.

The protection system is designed to allow online testing. Testing the protection system involves verification of the proper response, proper settings and proper operation of trip breakers. The impact on the availability of protection system signals is specific interest. That is, how the individual components of the protective functions are graded during test and maintenance activities.

Analog channels can be tested and maintained in either the bypassed or tripped state depending on the specific plant hardware capability. If tested in the bypassed state, the channel is unavailable. If tested in the tripped state, the channel is providing a trip signal to the logic. The logic, master relay and slave relays are tested and maintained in the bypassed state so these are unavailable during those activities. The reactor trip breakers are tested and maintained in the bypassed state, but the bypass trip breaker is used to provide reactor trip function from two breakers during the main trip breaker is tested or maintained. The undesirable aspect of this test is that a single failure or spurious signal on any redundant channel will cause a reactor trip thus subjecting the plant to a significant and unnecessary transient. Not only does this trip cause a thermal transient of the reactor and steam system components and piping, it challenges many of the systems important to plant shutdown.

TS currently require monthly testing of almost all RPS and ESFAS analog channels and bimonthly testing of the SSPS and trip breakers. The time allowed for testing analog channels, logic cabinets, master relays, slave relays , trip breakers by TS is 2 hours .The time allowed for maintenance of analog channel is 1hr and that of the others are 6 hours.

Ⅲ. The Analysis of Unavailabilities due to relaxing AOTs and STIs

To analyze the impact of increasing AOTs and STIs on system unavailability, a fault tree analysis of the individual reactor trip 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

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

, where $\lambda T < 0.1$. The unavailability, therefore 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 were modeled in the fault tree. THERP(Technique for Human Error Rate Prediction) method were 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 was 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 analog channel, the MGL approach was used

For the failure database of each component, plant specific data of Kori units 3,4 are obtained by using Bayesian update method for plant data and the Westinghouse data base, WCAP-10271. Table 1 lists the representative components failure data used in this analysis. It shows that the plant specific failure data of Kori units 3,4 have lower values than the generic Westinghouse data.

Fault tree were constructed to model the each signal of RPS/ESFAS to allow the calculation of the unavailability of individual trip functions. Each 17 RPS and 11 ESFAS signal was 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 are listed in Table 2 for evaluation of fault tree and core damage frequency.

Table 1. Plant specific failure data base of Kori units 3,4 .

Component	Generic Failure Rate(/hr)	Failure Data			Specfic Failure Rate(/hr)	Notes
		No of Failure	Operation hour	Failure Rate		
Pressure Transmitter	2.8E-6	3	4011059	7.48E-7	9.36E-7	
Temperature Transmitter	8.6E-6	0	776334	1.29E-6	2.52E-6	1
Level Transmitter	4.90E-6	3	2070224	1.45E-6	1.77E-6	
Channel Test Card	1.7E-7	1	17855682	5.6E-8	8.2E-8	
Loop Power Supply	5.8E-6	0	17855682	5.6E-8	2.26E-7	1
Signal Comparator Card	2.9E-6	4	28465508	1.41E-7	2.2E-7	
Lead/Lag Amplifier Card	7.8E-7	7	10868676	6.44E-7	6.24E-7	
Input Relay(SSPS)	8.7E-8	0	17855682	5.6E-8	6.16E-8	1
Undervoltage Output Card	-	0	1552668	6.44E-7	-	2
Master Relay	8.5E-6	1	2846558	3.51E-7	1.0E-6	
Slave Relay	8.5E-6	0	6340061	1.58E-7	5.47E-7	1

Notes 1 ; It is assumed that the failure occurred once when the failure did not occur before.

Notes 2 ; Analyzed by the component base

Table 2. The current and the proposed AOTs and STIs.

	Item	Current	Case 1	Case 2	Case 3
Analog Channel	Test Interval	1	3		
	Test Time	2	4	12	8
	Maintenance interval	24			
	Maintenance Time	1	12	78	30
Logic Cabinet	Test Interval	2			
	Test Time	2	4		
	Maintenance interval	18			
	Maintenance Time	6	12	30	18
Trip Breaker	Test Interval	2			
	Test Time	2			
	Maintenance interval	12			
	Maintenance Time	6			
Master Relay	Test Interval	2			
	Test Time	2	4		
	Maintenance interval	Failure Rate			
	Maintenance Time	6	12	30	18
Slave Relay	Test Interval	3			
	Test Time	2	4		
	Maintenance interval	Failure Rate			
	Maintenance Time	6	12	30	18

Time; Hour Rate, Interval : Month Rate

IV. Risk Analysis Results

The risk analysis is 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 NUPRA code to calculate core damage frequency. The base case was initially quantified with the signal unavailabilities corresponding to current AOTs and STIs. These were followed by quantifications with the signal unavailabilities for each case in Table 2. In addition to that, unnecessary plant transients and challenges to the protection systems caused by test were considered. An evaluation of CDF caused by forced outages that occurred from the commercial operation date to March, 1997 on K-3,4 plants, was performed. The core damage frequency and reactor trip risk during test were listed in table 3. As shown from Table 3, the increase in CDF of case 1 is 1.54%. These increases are relatively an insignificant impact on plant safety. The qualitative insights such as the efforts being taken to improve plants performance and to increase plant safety were also taken into consideration and looked up for the Kori 3,4 units. The examples of system upgrades and safety operation & maintenance efforts related to testing RPS/ESFAS were listed in Table 4. It is hard to quantify these effects on CDF, but from qualitative safety aspects, these are judged to increase plant safety as well as plant availability.

Table 3. Sensitivity study of core damage frequency.

Initiating Event	Current CDF	Case 1		Case 2		Case 3	
		CDF	%	CDF	%	CDF	%
LOCA, Transient	7.773E-5	7.887E-5	1.44	7.898E-5	1.58	7.898E-5	1.58
ATWS	1.27E-6	1.37E-6	0.13	1.64E-6	0.47	1.48E-6	0.27
Rx Trip Risk During Test		-2.14E-8	-0.03	-2.14E-8	-0.03	-2.14E-8	-0.03
Sum	7.9E-5	8.022E-5	1.54	8.060E-5	2.02	8.044E-5	1.82

Table 4. The examples of system upgrades and preventive maintenance not quantified to CDF.

Prevent Maintenance	Upgraded System
Visual Test	Voltage Tap adjustment in backup DC power
ICT(In Circuit Test)	Dual fuse in NCD card
7300 Function Tester ROMP(Repair Operation And Maintenance Program)	Separation of Power Source in NSSS 7300 Cabinet
Integrated performance test	Install Blower to reduce spurious signal caused by over heat

V. Conclusions

The impact on plant safety were analyzed in the case of changing the AOTs and STIs of RPS/ESFAS. The extended AOTs and STIs results in slight increase of CDF, but it can be considered as an negligible level. However, there are several benefits to plant safety and operation as follows : 1) preventing unnecessary plant transients and challenge 2) reducing a significant amount of time and attention on the part of the operation and 3) improving plant availability by reducing undesirable trip during test.

In conclusion, PSA results from relaxing AOTs and STIs show justification of revisions on TS.

References

1. Probabilistic risk analysis of the RPS and ESFAS test times and completion times,WCAP-14333-P,May 1995

2. Evaluation of Surveillance Frequencies and out of service times for the reactor protection instrumentation system, WCAP-10271, May 1989