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Reliability Analysis of the Reactor Protection System Using Markov Processes

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마코프 프로세스를 이용한 원자로 보호계통의 신뢰도 분석

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Abstract

The event tree/fault tree techniques used in the current probabilistic risk assessment (PRA) of nuclear power plants are based on the binary and static description of the components and the system. While these techniques may be adequate in most of the safety studies, more advanced techniques, e.g., the Markov reliability analysis, are required to accurately study such problems as the plant availability assessments and technical specifications evaluations that are becoming increasingly important.

This paper describes a Markov model for the Reactor Protection System of a pressurized water reactor and presents results of model evaluations for two testing policies in technical specifications.

요 약

현재 원자력발전소의 확률론적 위해도 평가에 사용되는 사상 수목이나 고장수목 기법은 부품이나 계통의 이원적상태와 정적 묘사에 근거하고 있다. 이 기법이 대부분의 안전해석에는 적합하지만, 요사이 점차 중요관심사가 되고 있는 발전소의 이용률 측정이나 기술 사양서 평가 같은 문제를 정확하게 다루기 위해서는 마코프 신뢰도 분석과 같은 보다 진보된 기법이 필요하다.

이 논문은 가압경수로의 원자로 보호계통을 위한 마코프 신뢰도 모델을 기술하고 기술사양서의 두 검사 절차를 분석한 결과를 제시한다.

I. Introduction

The main tools currently used in the probabilistic risk assessment (PRA) of the nuclear power plants are the event tree/fault tree techniques[1] These *event tree/fault tree models* are characterized by the ***binary*** and ***static*** descriptions of the components of the system under study. The binary description of a component assumes that it can be in two states, either in the up (operating) state or in the down (failure) state. The System state is also assumed to be binary.

The static description is equivalent to looking at the components and the *system at an instant* of time like taking a snap-shot, by averaging the characteristics (e.g., instantaneous unavailability) of a component over a time interval and then no longer considering time dependence of the system characteristics.

The binary and static descriptions are adequate in most of the safety assessments of nuclear power plants. However, they can be a limitation in some problems in safety analysis, and in problems such as plant availability studies and technical specifications evaluations where multiple states and stochastic behavior of the components and the system are important, they can be a serious limitation. The Markovian reliability analysis based on Markov processes is more appropriate for such problems. It is capable of handling multiple states and stochastic dependency inherent in the components and the system.

This paper describes the fundamentals of the Markov processes and the reliability analysis based on the Markov processes. The paper also presents some results of its applications to evaluating technical specifications of the Reactor Protection System (RPS) of a nuclear power plant.[2]

The paper is organized as follows. Section II briefly describes the Markov processes and the Markovian reliability analysis including a computer implementation. Section III provides the results of applications to the Reactor Protection System. Section IV presents the conclusions of the study.

II. Markov Processes

II.A. Definitions[3,4]

A discrete-state, continuous-time random process describes the stochastic behavior of a system that can be in any of a finite number (z) of discrete states and that changes its state randomly as a function of time. A change of state (state transition) can occur at any instant of time.

A discrete-state, continuous-time Markov Process is a random process such that the probability that the system will perform a state transition from state i to state j at any time depends only on the initial state i and the final state j of the transition and that the time until a change from one state to another occurs is an exponentially distributed random variable.

If $\pi_i(t)$ denotes the probability that the system is in state i at time t, and $\pi(t)$ the $1\times z$ row vector with elements $\underline{\pi}_i(t)$, for $i=1, 2, \cdots, z$, then it can be shown that $\underline{\pi}(t)$ satisfies the state evolution equation given by the relation

$$\dot{\underline{\pi}}(t)=\underline{\pi}(t)\underline{A} \qquad (2.1)$$

where $\underline{A}$ is a $z\times z$ matrix with elements a_{ij} such that:

$a_{ij}dt$=the probability that the system will transit to state j during the interval between t and $t+dt$ given that it is in state i at time t.

Vector $\underline{\pi}(t)$ is called the state probability vector with elements the state probabilities $\pi_i(t)$'s. Matrix $\underline{A}$ is called the transtion-rate matrix with elements the transition rates a_{ij}'s.

The solution of Eq. (2.1) is given by the relation

$$\underline{\pi}(t)=\underline{\pi}(0)\exp(\underline{A}t) \qquad (2.2)$$

where $\underline{\pi}(0)$ is the given value of the state probability vector at time $t=0$.

A *discrete-state, discrete-time Markov Process* is a random process such that: (a) state transitions occur at discrete times t_n, where

$$t_n=t_{n-1}+\Delta t(n),$$

or, with $\Delta t(n)$=constant,

$$t_n=t_0+n\Delta t;$$

and (*b*) the probability that the system will perform a state transition form state i to state j at time t_n depends only on the states i and j of the transition.

It can be shown that the $1\times z$ state probability vactor $\underline{\pi}(n)$ obeys the relation

$$\underline{\pi}(n+1)=\underline{\pi}(n)\underline{P}(n) \qquad (2.3)$$

where the $z\times z$ transition probability matrix $P(n)$ has, as elements, the transition probabilities $p_{ij}(n)$, for $i, j=1, 2, \cdots, z$.

II.B. Computer Implementation

The computer codes STAGEN and MAR-

ELA[2,5] can be used to quantify reliability models that are based on the Markov processes.

In the STAGEN code, component states are assigned to every component of the system. System states are then obtained by generating possible combinations of component states and grouped into system operating (X set) and system failed (Y set) groups. These sets are further divided into subgroups determined by the number of failed components.

The MARELA code receives, as input, the system states generated by STAGEN. In MARELA, additional system states can be defined depending on the interests of the user. For example, an additional syatem state will be the catastrophic state to which the system transits if there is a demand while it is in a state of Y set. In the case of the Reactor Protection System analysis, the system state of spurious scrams would be also considered important besides the catastrophic system state of anticipated transients without scram (ATWS).

MARELA generates the state-transition probability matrix $\underline{P}$ as a discretized approximation to the exponential expression of Eq. (2.2)

$$\underline{P}=\exp(\underline{A}\Delta t)\approx\underline{I}+\underline{A}\Delta t \quad (2.4)$$

for a sufficiently small time step Δt chosen in consideration of numerical accuracy and computation time.

Testing models 6 are also included in MARELA. A test-transition probability matrix $\underline{T}_k$ is defined for each component. If N is the number of components tested during a particular test, the test-transition probability matrix $\underline{T}$ is given by the relation

$$\underline{T}=\prod_{k=1}^{N}\underline{T}_k=\underline{T}_1\underline{T}_2\cdots\underline{T}_N \quad (2.5)$$

The matrices $\underline{P}$ and $\underline{T}$ are stored judiciously taking advantage of the sparseness of the structure to reduce computer storage requirement.

The MARELA code then solves recursively

$$\underline{\pi}(n+1)=\underline{\pi}(n)\underline{PQ}(n), \quad n=0,1,2,\cdots \quad (2.6)$$

in which a time-dependent matrix $\underline{Q}(n)$ is defined as

$$\underline{Q}(n)=\underline{T\Delta}(n-kT)$$

where $\underline{\Delta}$ is a matrix with elements

$$\delta_{ij}(n-kT)=\begin{cases}1 \text{ if } i=j \text{ and } n=kT,\ k=1,2,\cdots\\ 0 \text{ otherwise}\end{cases} \quad (2.7)$$

to allow for the fact that the test is periodically repeated every T time steps.

At each time step n, the time-dependent unavailability is calculated by the relation

$$U(n)=\sum_{i\in Y}\pi_i(n) \quad (2.8)$$

and the average unavailability over a period of n_0 time steps by the relation

$$\bar{U}=\frac{1}{n_0}\sum_{n=1}^{n_0}U(n) \quad (2.9)$$

The catastrophic failure probability and the addititonal state probabilities are directly calculated as the state probabilities of the corresponding states from Eq. (2.6)

III. Applications

III. A. Markov Model of the Reactor Protection System (RPS)

This section describes briefly the Markov model developed for the electrical portion of the RPS[2]. The RPS is represented in a functional block configuration (Fig. 1). There are four blocks for analog channels and two for logic trains and trip breakers. Each functional block is considered as a supercomponent composed of several basic components in series.

The state transition diagram for an analog channel is given in Figure 2a. An analog channel is represented by a five-state component.

State 1: is the operating state.

State 2: is the the failed state. In this state the component is failed, the failure can be detected in the next test and the component will be put under repair.

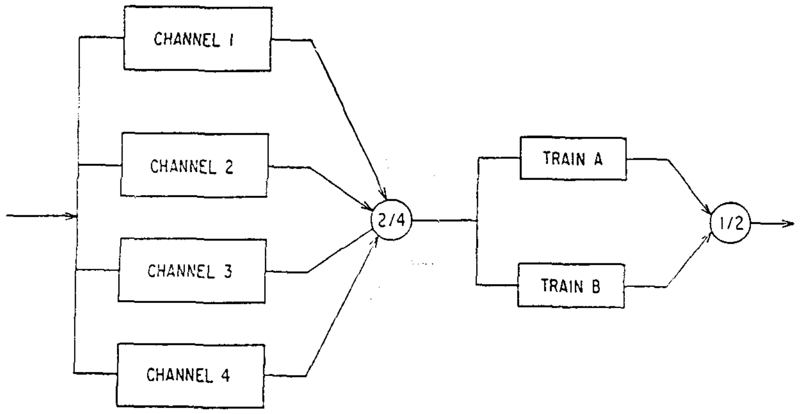

Fig. 1. Reactor Protection System Functional Block Diagram

State 3: is the tripped state. While in this state the channel generates a trip signal and it may undergo repair.

State 4: is the bypass state related to state 1. To perform a test the channel can be bypassed for a prespecified period of time: *Allowable Bypass Time* (τ). If the test is completed within the Allowable Bypass Time, the state of the component transits to state 2 or state 1 depending on whether there was a human error during the test or not. If the test is not completed within the Allowable Bypass Time, the component is put under state 3 at the end of the Allowable Bypass Time.

State 5: is the bapass state related to state 2. If the channel is failed the testing and repairing can be performed while in a bypass mode, provided that the Allowable Bypass Time (τ) is not exceeded.

When the allowable bypass time is small compared to the mean time of channel failure, the two test states (4 and 5) can be omitted by assuming that transitions in and out of states 4 and 5 occur instantaneously at the time of testing and with appropriate probabilities (see Fig. 2b).

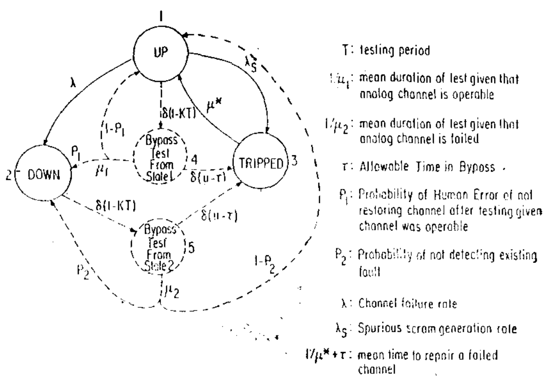

Fig. 2a. State Transition Diagram: Analog Channel, "Nonmarkovian Model"

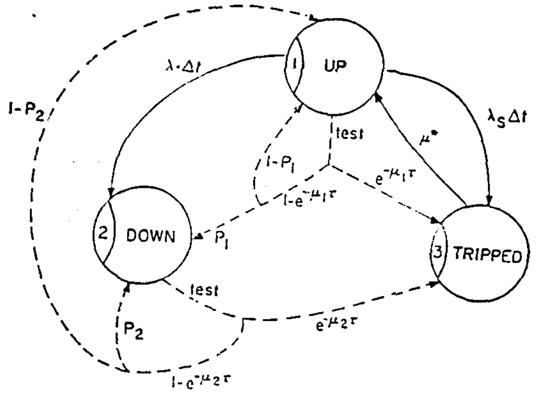

Fig. 2b. State Transition Diagram: Analog Channel, "Equivalent" Markovian Model

The state transition diagram for the logic train and trip breaker is similar to the one of the analog channel.

Fig. 3 shows the system-level state transition

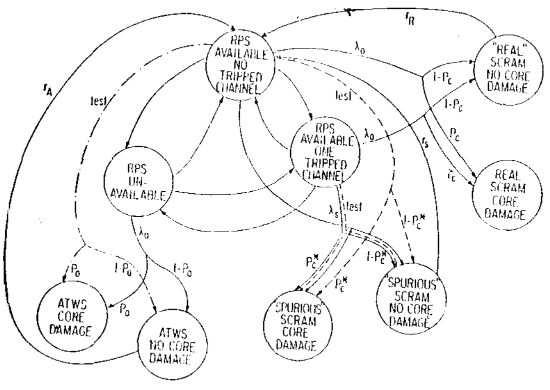

Fig. 3. Generalized State Transition Diagram: Reactor Protection System

diagram. The 198 system states generated by the computer code STAGEN can be grouped into the following nine groups:

1) *RPS Available With No Tripped Analog Channel:* This group contains all system states with at least two analog channels and one logic train operable.
2) *RPS Available With No Tripped Analog Channel:* This group contains all system states with one analog channel tripped and at least one more analog channel and one logic train operable.
3) *RPS Unavailable:* This group contains all the states that imply system unavailability (two logic trains or three analog channels failed).
4) *"Real" Scram No-Core Damage:* This group contains all the states of the system that imply an available RPS and the successful reactor shutdown following a "real" scram signal. Real signal means a signal generated by the RPS by properly responding to abnormal conditions of the plant.
5) *"Real" Scram-Core Damage*: This group contains all the system states that imply an available RPS and the reactor in core-damaged state. The RPS successfully responded to the "real" challenge but the decay heat removal function failed.
6) *"Spurious" Scram-No Core Damage:* This corresponds to Group No. 4 with the scram signal spuriously generated internally to the RPS.
7) *"Spurious" Scram-Core Damage:* This corresponds to Group No. 5 with a spurious scram initiator.
8) *ATWS-No Core Damage:* This group contains all the system states that imply an unavailable RPS coupled with a real challenge (Anticipated Transient Without Scram-ATWS) but with successful mitigation of the event.
9) *ATWS- Core Damage*: This group contains all the system states that imply an unavailable RPS coupled with a real challenge (ATWS)that results in core damage.

Additional features of the model are staggered testing and inclusion of common-cause failures. The specific areas that the Markov model improves over the current PRA technique (e.g., fault tree analysis) are the following:

(i) *Modeling of Multiple States:* A component can be in any number of discrete states. In particular, the Markov model allows for the modeling of *Bypass* and *Trip* states for the analog channels and the *logic trains.* A current PRA technique would assume only one failed state (component unavailable) and it would assume that the component is unavailable every time it is tested and for a period of time equal to the mean time of the maintenance activity. This approach creates three problems:

(a) It introduces a conservatism in the calculation by overestimating the unavailability of the system. This is because when a channel is in a trip mode it takes three additional failures for the system to be unavailable. Assuming that the channel is unavailable, however, requires only two additional failures to fail the system.
(b) It introduces a nonconservatism by under-

estimating the probability of spurious scrams. When a channel is in a trip mode an additional spurious trip in any of the remaining channels will cause a spurious reactor scram.

(c) It introducesa difficulty in estimating the real effect of a policy change in limiting conditions of operation (LCO). It is conceivable that two alternative LCO policies are characterized by the *same* mean time to test and repair a channel and different allowable times in bypass.

(ii) *State Dependences:* The stochastic behavior of the system might depend on its state. For example, the allowable bypass time for an analog channel depends on whether another channel is already tripped or not. The repair rate of an analog channel might depend on whether another channel is under repair or on whether the reactor is shutdown or online. Exceeding the allowable bypass time in an analog channel will generate a scram signal depending on whether another channel is tripped or not and on whether the reactor is online or not.

(iii) *Renewal Effect of Challenges:* A successful challenge to the system will reveal any existing failures which will be subsequently repaired. Thus, the challenges to the system have the effect of randomly occuring tests. However, whether a challenge will have the equivalent effect of a test on a component will depend on whether the system is available at the time of the challenge.

(iv) *Inclusion of the "NO CORE DAMAGE" and "CORE DAMAGE" States:* The inclusion of no core damage states is important because they allow for the estimation of the expected reactor downtime that is directly related to the RPS. This quantity is an important attribute of any LCO policy. In addition, the inclusion of the no core damage and core damage states permit a more accurate estimation of the system unavailability and failure probability. This is due to the fact that the system spends a finite amount of time in the "no core damage states". The time the system spends in states of Groups 1 to 3 is then reduced accordingly and thus some double counting in the estimation of the systems unavailability and failure probability is avoided.

The Markov model calculates the effect of these characteristics by considering their impact *dynamically,* that is, as a function of time.

III. B. Results and Discussions

The Markov model described in Section III. A was quantified using the data base given in Tables 1 and 2. Table 1 shows the failure rates of the components comprising the analog channels, the logic trains, and the trip breakers. Table 2 shows the repair data and other parameters required in the model.

Two attributes are of interest:

(1) the probability of core damage per year of reactor operation and (2) the average reactor down time per year of reactor operation.

The above two attributes were calculated for two testing policies summarized in Table 3. Policy 1 is the current limiting conditions of operation (LCO): allowable bypass times are 1, 2, and 1 hour for one chanel, second channel if another is tripped, and a logic train, respectively; 30 days for the channel test interval; 60 days for the logic train test interval. Policy 2 is the proposed LCO: 6, 4, 4 hours and 90, 180 days, correspondingly.

Since some of the parameters, i.e., β-factors and the core damage probability given an ATWS (CORED/ATWS), are considered uncertain to a large extent, sensitivity studies were performed for a range of values for these parameters.

The core damage probability given a scram (CORED/SCRAM) was evaluated at 1. 43E-5 in this study by using the Indial Point-3 PRA as revised by Sandia National Laboratories[11] (the core damage frequency resulting from all internal

Table 1. Failure Data

Component	Failure Mode	Failure Probability	Source	Comments
Analog Channel Block				
Input Relay	Fails to open	5.09(−7)/d	Ref. 7	5.81(−10)/hr*
	Operates spuriously	3.6 (−8)/hr	Ref. 7	
Loop Power Supply (120 VAC)	Inoperable[Δ]	5.4 (−7)/hr	Ref. 8	W-PWR data
	Reduced Capability[Δ]	9.1 (−8)/hr	Ref. 8	W-PWR data
Signal Conditioning Module	Inoperable[Ω]	2.6 (−6)/hr	Ref. 8	W-PWR data
	Reduced Capability[Ω]	1.55(−6)/hr	Ref. 8	W-PWR data
Comparator (Bistable)	Inoperable	6.5 (−7)/hr	Ref. 8	W-PWR data
	Reduced Capability	8.4 (−7)/hr	Ref. 8	W-PWR data
Sensor/Transmitter				
Neutron Flux	Inoperable	3.4 (−6)/hr	Ref. 8	W-PWR data
	Reduced Capability	8.5 (−7)/hr	Ref. 8	W-PWR data
Pressure	Inoperable	2.6 (−7)/hr	Ref. 8	W-PWR data
	Reduced Capability	3.1 (−6)/hr	Ref. 8	W-PWR data
Total				
Flux Channel	Fails to operate	6.65(−6)/hr		
	Operates spuriously	3.91(−6)/hr		
Pressure Channel	Fails to operate	3.51(−6)/hr		
	Operates spuriously	6.16(−6)/hr		
Logic Train and Trip Breaker Block				
Trip Breaker	Fails to open	2.27(−4)/d	Ref. 7	2.59(−7)/hr*
	Operates spuriously	4.3 (−8)/hr	Ref. 7	from indoor AC breakers
UV Coils	Fails to open	5.09(−7)/d	Ref. 7	5.81(−10)/hr*
	Operates spuriously	3.6 (−8)/hr	Ref. 7	from relay coils
DC Power (48V) for UV Coils	Inoperable	5.4 (−7)/hr	Ref. 8	W-PWR data
	Reduced Capability	9.1 (−8)/hr	Ref. 8	W-PWR data
Solid State Logic Circuits	Fails to operate	1.73(−6)/hr	Ref. 7	from solid
	Operates spuriously	2.48(−6)/hr	Ref. 7	state alarms
DC Power (15V) for Solid	Inoperable	5.4 (−7)/hr	Ref. 8	W-PWR data
State Logic Circuits	Reduced Capability	9.1 (−8)/hr	Ref. 8	W-PWR data
Total	Fails to operate	2.52(−6)/hr		
	Operates spuriously	3.28(−6)/hr		

* Converted to an hourly failure rate assuming 10 demands per year.

Δ Both failure modes of power supply are considered to produce spurious signals.

Ω In Ref. 8 "Inoperable" is defined as failure events involving actual failure and "Reduced Capability" as instrument drift, out-of-calibration, intermittent (spurious) events. The condition of reduced capability is considered to produce spurious signals.

transients was divided by the total frequency of internal transient initiators).

Six cases defined by the parameters shown in Table 4. were evaluated using CORED/SCRAM =1.43E-5 and the results are presented in Table 5. Core damage frequencies are provided in Table 6. and in Figure 4 for several values of assumed CORED/ATWS and β-factors.

It is observed from the results that whether Policy 2 represents an improvement over Policy 1 depends on CORED/ATWS, as well as, on the degree of dependence between the channels

Table 2. Data for Parameters of the Model

Parameter	Data (Mean)	Source	Comments
μ_1^{TR}	$1hr^{-1}$	Ref. 9	
μ_2^{TR}	$1/7hr^{-1}$	Ref. 9	
μ_1^{CH}	$1hr^{-1}$	Ref. 9	
μ_2^{CH}	$1/7hr^{-1}$	Ref. 9	
μ_{31}^{CH}	$1/16hr^{-1}$	Ref. 9	
λ_0	$9.91yr^{-1}$	Ref. 10	Challenge rate on RPS (Frequency of transients)
$r_s=r_R$	25.6hrs	Ref. 9	
P_C	1.43E-5/d	Ref. 11	Indian Point-3 PRA revised by Sandi(Intenal transient initiators)
	1.60E-6	Ref. 12	Zion PRA (Internal initiators)
P_c*	5.21E-7	·Ref. 11	
P_0	6.42E-2	Ref. 11	

Table 3. Testing Schedules (Limiting Conditions of Operation) **Considered in the Study***

Policy	T^{CH}(Days)	T^{TR}(Days)	τ(Hours)	τ_0(Hours)	τ_1(Hours)
1	30	60	1	2	1
2	90	180	6	4	4

* T^{CH}, T^{TR} : Test intervals for channels and logic trains, respectively.

τ : Allowable bypass time for an analog channel test.

τ_0 : Allowable bypass time for an additional analog channel test if one is already tripped.

τ_1 : Allowable bypass time for a logic train test.

Table 4. β-Factors and Limiting Conditions of Operation

Case	β^{CH}	τ_{β}^{R}	T^{CH}(Days)	T^{TR} (Days)	τ(Hours)	τ_0(Hours)	τ_1(Hours)	
1	0.1	0.1	30	60	1	2	1	Current LCO
2	0.1	0.1	90	180	6	4	4	Proposed LCO
3	0.01	0.01	30	60	1	2	1	Current LCO
4	0.01	0.01	90	180	6	4	4	Proposed LCO
5	0.	0.	30	60	1	2	1	Current LCO
6	0.	0.	90	180	6	4	4	Proposed LCO

β : β-factor

T : Test ilnterva

τ : Allowable bypass timefor an analog channel test

τ_0 Allowable bypass timefor an additional analog channel test if one channel is already tripped.

τ_1 Allowable bypass time for a logic train and trip breaker test.

and between the trains (β-factors). In the absence of dependences (β=0), Policy 2 results in a lower core damage frequency for almost all the values of CORED/ATWS. For relatively weak dependences (β=0.01), Policy 2 is better than policy 1 only for values of CORED/ATWS less than 0.2. For relatively strong dependences (β=0.1), Policy 2 is better than Policy 1 only for values of CORED/ATWS less than 0.025.

Figure 5 shows in CORED/ATWS-β plane a limit curve which divides the plane into two regions where one testing policy is better than the other and vice versa.

The six cases were repeated using a different CORED/SCRAM (=1.60E-6) estimated from the Zion Probabilistic Safety Study[12] and the

Table 5. Results of the Markov Model(CORED/SCRAM=1.43E-5)

Case	Average Unavailability	ATWS	Average Real Scram	Core Damage Due to Real Scram	Average Spurious Scram	Core Damage Due to Spurious Scram
1	2.040(−4)	1.801(−3)	2.718(−2)	1.224(−4)	9.773(−3)	5.793(−5)
2	4.594(−4)	4.126(−3)	2.749(−2)	1.237(−4)	3.947(−4)	2.113(−6)
3	2.140(−5)	2.018(−4)	2.720(−2)	1.225(−4)	9.781(−3)	5.797(−5)
4	4.988(−5)	4.807(−4)	2.754(−2)	1.240(−4)	3.954(−4)	2.117(−6)
5	1.095(−6)	2.401(−5)	2.721(−2)	1.225(−4)	9.782(−3)	5.798(−5)
6	4.279(−6)	7.474(−5)	2.755(−2)	1.240(−4)	3.955(−4)	2.118(−6)

Table 6. Probability of Core Damage per Reactor Year(CORED/SCRAM=1.43(−5))

Conditional Probability of Core Damage Given ATWS	β=0.		β=0.01		β=0.1	
	Policy 1	Policy 2	Policy 1	Policy 2	Policy 1	Policy 2
10^{-3}	1.81(−4)	1.26(−4)	1.81(−4)	1.27(−4)	1.82(−4)	1.30(−4)
5×10^{-3}	1.81(−4)	1.26(−4)	1.8 (−4)	1.29(−4)	1.89(−4)	1.46(−4)
10^{-2}	1.81(−4)	1.27(−4)	1.83(−4)	1.31(−4)	1.98(−4)	1.67(−4)
5×10^{-2}	1.82(−4)	1.30(−4)	1.91(−4)	1.50(−4)	2.70(−4)	3.32(−4)
10^{-1}	1.83(−4)	1.34(−4)	2.01(−4)	1.74(−4)	3.69(−4)	5.38(−4)
5×10^{-1}	1.92(−4)	1.63(−4)	2.81(−4)	3.66(−4)	1.08(−3)	2.19(−3)
1.	2.04(−4)	2.01(−4)	3.82(−4)	6.07(−4)	1.98(−3)	4.25(−3)

Table 7. Limit Points in CORED/ATWS vs. β-Factors

Conditional Probability of Core Damage Given Scram CORED/SCRAM		Limit Points	
	β=0.	β=0.01	β=0.1
1.43(−5)	1.0	2.0(−1)	2.5(−2)
1.60(−6)	1.2(−1)	2.2(−2)	2.6(−3)

Table 8. Limit Points in Ratios of CORED/ATWS to CORED/SCRAM vs. β-Factors

Conditional Probability of Core Damage Given Scram CORED/SCRAM		Limit Points	
	β=0.	β=0.01	β=0.1
1.43(−5)	6.99(4)	1.40(4)	1.75(3)
1.60(−6)	7.50(4)	1.38(4)	1.63(3)

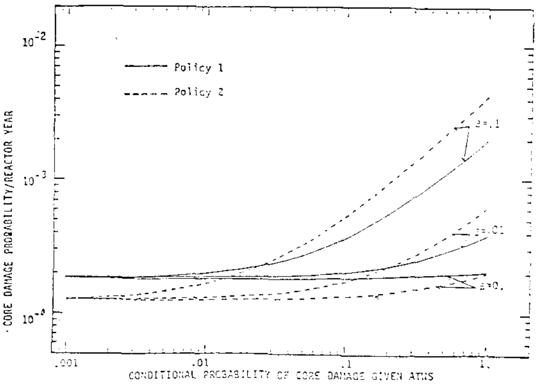

Fig. 4. Core Damage Probability Per Reactor Year (CORED/SCRAM=1.43×10^{-5}).

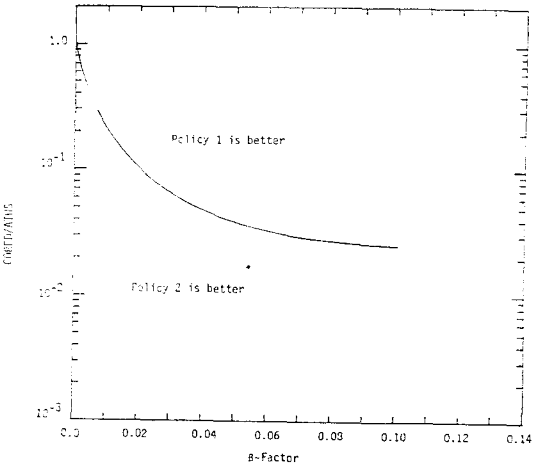

Fig. 5. A Limit Curve in CORED/ATWS-β Plane (CORED/SCRAM=1.43E-05)

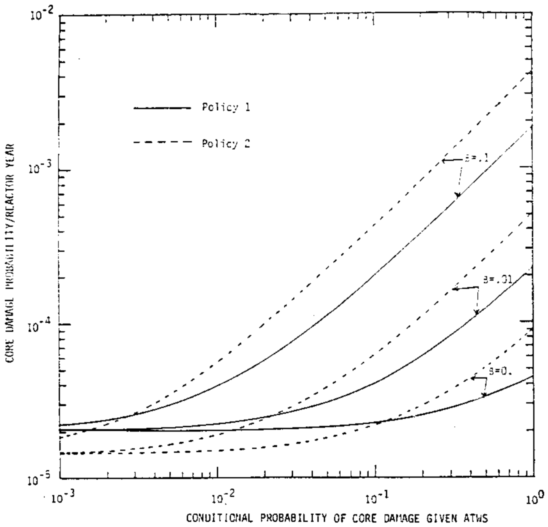

Fig. 6. Core Damage Probability per Reactor Year (CORED/SCRAM=1.60E-6)

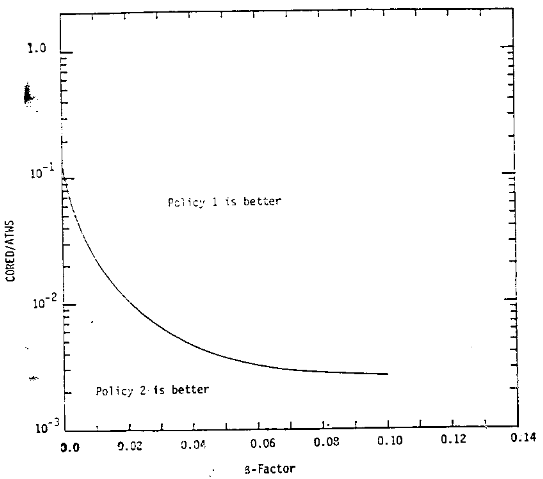

Fig. 7. A Limit Curve in CORED-ATWS-β Plane (CORED/SCRAM=1.60E-06)

otherwise same parameters given in Table 4. The results are shown in Figures 6 and 7.

Table 7 shows for the two values of CORED/SCRAM, the limit points in terms of CORED/ATWS which were read from the crosspoints in Figures 4 and 6. These limits points were plotted in Figures 5 and 7. It can be see by comparing Figures 5 and 7 that the region where Policy 2 is better than Policy 1 is smaller if the CORED/SCRAM is smaller. In other words, for a plant whose CORED/SCRAM is estimated to be small (e.g., by a PRA), the β-factors and/or the CORED/ATWS of the plant should be also small for the Policy 2 to be competitive with and better than Policy 1.

Table 8. shows the same limit points, but now in terms of the ratio of CORED/ATWS to CORED/SCRAM, and the limit points are also shown in Figure 8. It is interesting to note that the two limit curves lie very close together and are essentially indistinguishable. This may imply that the ratio of CORED/ATWS to CORED/SCRAM, not the individual parameters, is an important factor in deciding which testing policy is better.

It is thus recognized that decision making on LCO policies of the RPS will become much easier if there exists a plant-specific PRA which provides more definitive and plant-specific parameters such as CORED/SCRAM and CORED/ATWS.

Figure 9 shows for Policy 1, the core damage probability, as a function of time, resulting from three contributors: (1) core damage due to

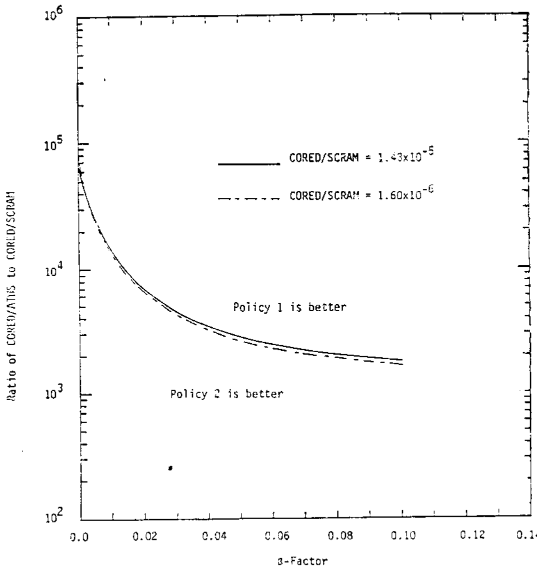

Fig. 8. Limit Curves in CORED-ATWS/CORED-SCRAM-β Plane

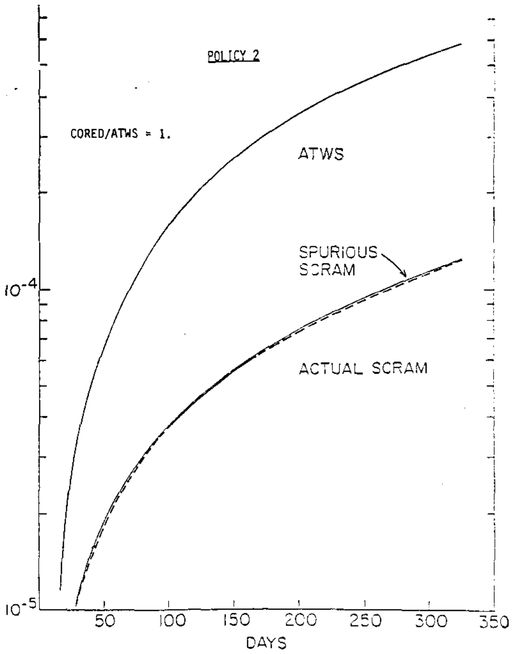

Fig. 9. Contributors to Core Damage Probability as a Function of Time-Policy 1

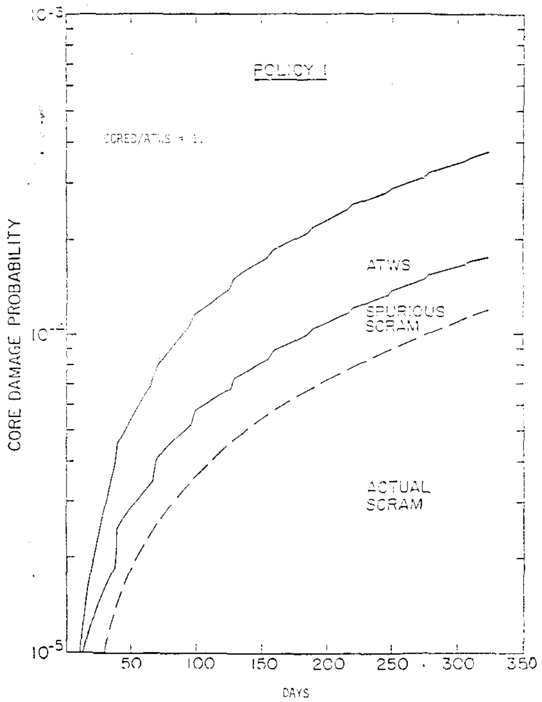

Fig. 10. Contributors to Core Damage Probability as a Function of Time-Policy 2

Table 9. Comparison of Two Policies

	Core Damage Probability/RY	Reactor Down Time (Fraction)
Policy 1	1.31(−4)	3.50(−2)
Policy 2	1.67(−4)	3.11(−2)

ATWS (assuming a particular value of CORED/ATWS=1), (2) core damage due to spurious scrams, and (3) core damage due to real scrams.

The core damage probability from spurious scrams clearly shows a periodic time behavior. However, it appears that the other two contributors and total core damage probability do not show periodicity. Actually they also show similar periodicity but they are not as pronounced as the spurious scrams in the figure. This can be explained by the fact that the spurious scram probability exhibits much larger variations than the unavailability[2].

Figure 10 shows similar information for Policy 2. Contribution from ATWS becomes larger and contribution from spurious scram becomes substantially smaller compared to Policy 1.

Table 9. summarizes the results of calculation of the two attributes for Policy 1 and Policy 2.

IV. Conclusions

The binary and static modeling in the event tree/fault tree techniques can be a serious limitation in problems such as plant availability studies and technical specifications evaluations. The Markov model is more appropriate for such problem where multiple states and stochastic behavior of the components and the system are important.

A Markov model was developed for the Reactor Protection System of a pressurized water reactor and evaluated for the two specific testing policies (two sets of technical specifications). The model calculated two attributes, i.e., core damage probability (CDP) per year of reactor operation and average reactor downtime (ARD) per year of reactor operation.

The point value calculations show that the change from Policy 1 (current LCO) to Policy 2 (relaxed LCO) results in an increase of CDP by 27 percent ("impact") and in a decrease of ARD by 11 percent ("value"). Thus, Policy 1 is preferred to Policy 2 if the core damage probability is the sole attribute of performance, while Policy 2 is preferred to Policy 1 if the reactor downtime is the sole attribute of performance. Deciding on one policy against the other requires a decision maker's value tradeoffs between the attributes of performance.

General conclusions based on several sensitivity calculations are as follows:

1. The CDP is mainly affected by the unavailability of the RPS and consequently by the probability of an ATWS. The ARD, however, is mainly affected by the probability of spurious scrams. This behavior is due to the fact that the conditional probability of core damage given an ATWS is much higher than the conditional probability of core damage given a spurious scram.

2. The ABTs for the analog channels and the logic trains affect mainly the probability of spurious scrams. In general, an increase in the ABTs results in a decrease in the probability of a spurious scram and in a much smaller increase in the probability of an ATWS. The conditional probability of core damage given a spurious scram is, however, much smaller than the conditional probability of core damage given an ATWS. Consequently, an increase of the ABTs results in a small net increase of the CDP. On the other hand, the significant decrease in the probability of spurious scrams corresponds to a significant decrease in the ARD. This behavior is observed for all levels of dependences among the analog channels and between the logic trains (dependent failures) and it is slso supported by the uncertainty analysis[2].

3. The test intervals of the analog channels and of the logic train affect the CDP more than they affect the ARD, and they do so in a way that depends on the level of dependences among the analog channels and between the logic trains (dependent failures). At low levels of dependences large test intervals are justified, while at high levels of dependences small test intervals result in lower CDP.

References

1. "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, NUREG-75/014 (WASH-1400), October 1975.
2. Papazoglou, I.A. and Cho, N.Z., "Review and Assessment of Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," Brookhaven National Laboratory, BNL-NUREG-51780, April, 1984.
3. Howard, R., "Dynamic Probabilistic Systems," Volumes I and II, John Wiley and Sons, Inc., New York, 1971.
4. Shooman, M.D., "Probabilistic Reliability: An Engineering Approach," McGraw-Hill Book Company, New York, 1968.
5. Papazoglou, I.A. and Gyftopoulos, E.P., "Markov Processes for Reliability Analyses of Large Systems," IEEE Trans. Reliability, R-26, 232 (1977).
6. Papazoglou, I.A. and Sun, Y.H., "Risk Evaluation of Generic Fluid Systems," Brookhaven National Laboratory, NUREG/CR-3528, Review Draft, July, 1982.
7. IEEE Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability Data for Nuclear Power Generating Station, IEEE Std., 500-1977.
8. Miller, C.F. et al., "Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants," EG & G Idaho, Inc., NUREG/

CR-1740, May 1981.
9. Jansen, R.L., Lijewski, L.M., and Masarik, R.J., "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," Westinghouse Electric Corporation, WCAP-10271, January 1983.
10. McClymont, A.S. and Poehlman, B.W., "ATWS: A Reappraisal Part 3: Frequency of Anticipated Transients," Electric Power Research Institute, EPRI NP-2230, January 1982.
11. Kolb, G.J. et al., "Review and Evaluation of the Indian Point Probabilistic Safety Study," Sandia National Laboratories, NUREG/CR-2934, December, 1982.
12. Zion Probabilistic Safety Study, Commonwealth Edison Company, September 1981.