A STOCHASTIC RELIABILITY MODEL FOR NUCLEAR POWER PLANT SAFETY SYSTEMS CONSIDERING THE QUALIFIED LIFE EXTENSION AND THE MAINTENANCE EFFECTIVENESS MONITORING PROGRAM

N. J. Ferro, newtonferro@yahoo.com.br
COPPE/UFRJ – Programa de Engenharia Nuclear, Caixa Postal 68509, CEP 21941-972 Rio de Janeiro, RJ, Brasil.

P. L. C. Saldanha, saldanha@cnen.gov.br
CNEN-CGRC – C. R. General Severiano 90, sala 426, CEP 22290-901 Rio de Janeiro, RJ, Brasil.

P. F. Frutuoso e Melo, frutuoso@con.ufrj.br
COPPE/UFRJ – Programa de Engenharia Nuclear, Caixa Postal 68509, CEP 21941-972 Rio de Janeiro, RJ, Brasil.

Abstract. *This paper presents a model that can be implemented in a personal computer, which allows for the reliability analysis of safety systems of nuclear power plants, starting from a state transition diagram, typically employed in Markovian reliability analyses. Due to the need of considering the equipment qualified life extension, an issue of utmost importance in nuclear power plants nowadays, the classical Markovian model will not apply for at least one of the transition rates in the state transition diagram is generally time-dependent, due to equipment aging. To overcome this difficulty, the model is recast into a Markovian model by means of supplementary variables. As nuclear power plants are subject also to the so called maintenance rule, a requirement set up by the nuclear regulatory body, the utility needs to develop and present a maintenance program which is termed the Maintenance Effectiveness Monitoring Program in Brazil. The key feature here is the definition of maintenance procedures in the context of the maintenance program that are also adequate for the qualified like extension license. The algorithm developed for this purpose also estimates the necessary parameters for the reliability analysis, starting from crude failure information available from the plant and using goodness of fit statistical tests for the most commonly employed probability distributions, like the Weibull distribution. Once the transition rates are estimated, then a set of differential equations is solved by finite difference techniques. The algorithm is employed for analyzing data on a typical auxiliary feed-water system of a PWR plant. The obtained results can help making decisions on the maintenance program to be set and also on the qualified life extension. The developed model can also be useful for aging probabilistic safety assessments of nuclear power plants.*

Keywords*: Qualified life extension, Maintenance effectiveness monitoring program, Probabilistic safety assessment, Supplementary variables.*

1. INTRODUCTION

The evaluation of aging effects on the reliability figures of safety systems of a nuclear power plant is important before the implementation of the qualified life extension program. The demand for cost × benefit optimization of the resources applied in the maintenance of repairable components of these plants, without neglecting the safety and environmental aspects, leads to an unequivocal need for obtaining proposals that make it possible the accomplishment attendance of this qualification needs. The definition of tools that make it possible to monitor the maintenance effectiveness has been a general concern in the nuclear area.

2. THE MAINTENANCE RULE

The Maintenance Rule (MR) is a procedure established by the U. S. Nuclear Regulatory Commission (NRC) to check the effectiveness of the maintenance carried out in the nuclear plants under its responsibility. The MR technique was consolidated in the U.S. in 1996 and it is currently underway the discussion of the feasibility of its introduction in Europe and in Brazil.

The MR classifies Structures, System or Components (SSC) into two categories (NEI, 2000), the ones in Category (a) are evaluated as to the fulfillment of the SSC intended performance, while those in Category (b) are evaluated as to their relevance in relation to the plant safety. In summary, the MR procedure monitors the SSC as follows: Category (a)(2), the SSCs that reach the intended performance demonstrate that the preventive maintenance is being appropriately performed, and category (a)(1), which stands for the SSCs that do not fulfill category (a)(1), and must have established goals, so that the discrepancies can be revised and then return to Category (a)(2). The MR simplified flowchart can be found in NRC (2009).

The procedure establishes that all the SSCs can be evaluated to verify the pertinence of their inclusion in a MR. If the SSC is related directly to safety, can mitigate accidents or transients, is part of Emergency Operational Procedures (EOP), can prevent other SSC of performing their safety functions, or causes a reactor shutdown or a safety system actuation, the SSC will be put within the scope of the MR. Otherwise, it remains under the existing maintenance program, outside the scope of the MR.

An Expert Panel determines the relationship between the selected SSC and the plant safety within the MR scope. In the case of high or low significance, but in standby operation, criteria performance must be established on a train level, whereas for the SSCs in normal operation with low significance level, the criteria are established on a plant level. The inclusion of standby SSCs on a train level, even though they have a low degree of involvement with safety, is due to the fact that the failure of most standby systems can only be observed during their respective surveillance tests. In this sense, as plant transients occur with a low frequency, failure on demand data by itself could not provide the necessary information concerning the SSC monitoring and it is not a good indicator or measurement of the SSC performance. At last, SSCs are monitored to verify whether their performance meets the established criteria or whether they undergo a Maintenance Preventable Functional Failure (MPFF).

A periodic evaluation of the reliability and unavailability balance is scheduled, according to the MR, at each refueling cycle or every two years.

3. A COMPLEMENTARY APPROACH FOR THE MAINTENANCE RULE

When implementing the MR, the NRC observed that many licensees did not have an adequate technical basis to demonstrate that their goals and performance criteria were established with the necessary safety level, as long as they did not statistically demonstrate their association with the plant safety probabilistic analysis. Moreover, many licensees did not develop reliability and availability goals or performance criteria for safety significant SSCs as part of periodic evaluations (NRC, 1999).

To fill such gap, this work initially established a new approach to complement the MR, with the intermediate parameters insertion to be used as a tool to support the decision on the occasion of the inspection frequency previously established, or at any time if necessary. Such approach will also provide the possibility of using more realistic policies by means of the plant failure data. Thus, the algorithm is intended to consolidate and process the failure data, as they occur, so that they can be used in the areas responsible for the respective analyses and inserted in the context of the MR.

The complementary approach inserts the reliability parameters to be compared with the established performance criteria. As long as the parameters and criteria are compatible, that is, both approve or disapprove the monitored SSC, the regular periodic reevaluation of the SSC each 24 months or at the end of the refueling cycle is kept, as established in the procedure of the MR. When the parameters and criteria are incompatible, the feedback of the MR procedure begins, with the check of the algorithm that calculated the reliability parameters, being this algorithm revised whenever any discrepancy is identified. On the other hand, the performance parameter must be revised and corrected, if necessary. Figure 1 presents the new approach simplified flowchart.

4. PROPOSED MODEL PRESENTATION

The proposed model allows, starting from the failure data records, to perform a dynamic process analysis by means of a state transition diagram that allows for representing the system failure and success states and analyzing it through the Markovian reliability approach, using supplementary variables (Cox e Miller, 1965) to take into account possible time-dependent transition rates.

The Pareto's method is suggested in order to choose the failure processes to be analyzed. The necessary data for the sample should be generated by Maintenance Service Orders, which are censored to the right, since not all sample components will necessarily fail, implying that their failure occurs after the time period considered.

The data are organized by failure modes for the selected item and they are used to obtain the appropriate theoretical distribution for describing the component failure process through the parametric method, for which the exponential and the Weibull distributions are initially tested for their goodness of fit to the available data.

After the statistical treatment and the identification of the necessary parameters from the sample, a transition state diagram is used, where the transitions with time dependent rates (for failures data, repair and others) will appear. In this case, when the SSC presents aging characteristics, the boundary conditions are typically established, to organize a data base and make the modeling possible.

These conditions have to do with repair and/or replacement policies used in nuclear plants. The applicable technical specifications will be respected, when necessary. The information presented in the transition diagram is then cast into a coupled differential equations hybrid system whose solution will be through finite differences, using the data already mentioned. The hybrid character is due to the existence of both ordinary and partial equations. From the solution of these equations the reliability figures of interest are calculated as for example, the system unavailability.

In complement, it is intended that this model, after its validation, allows to be used not only for aging processes but, also, in their cycle of useful life, because both plants and components that are on their useful life are have failure times that follow the exponential distribution, which is a particular case of the Weibull distribution.

The proposed methodology is summarized in figure 2.

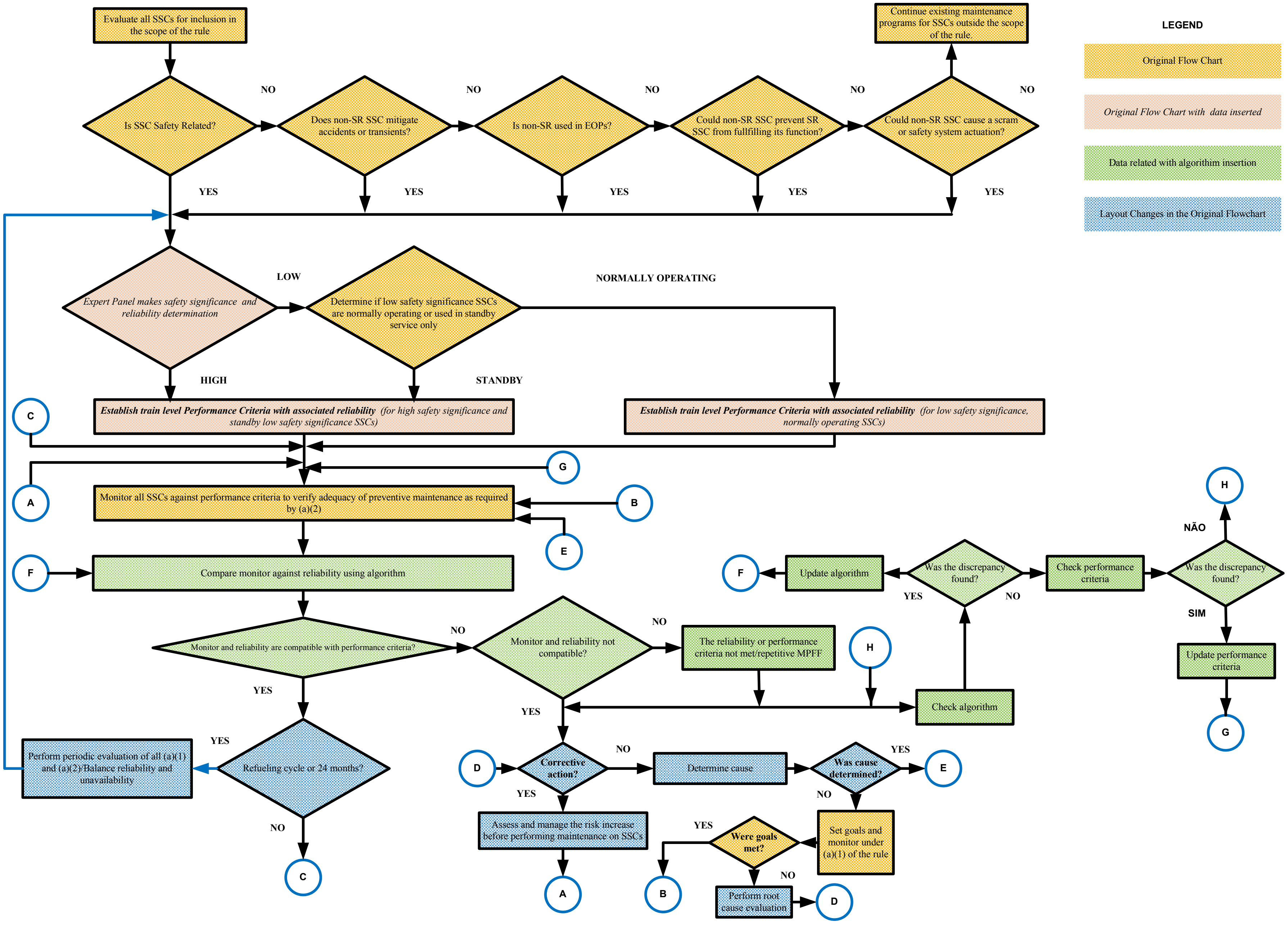

Figure 1 – New Approach Simplified Maintenance Rule Flow Chart

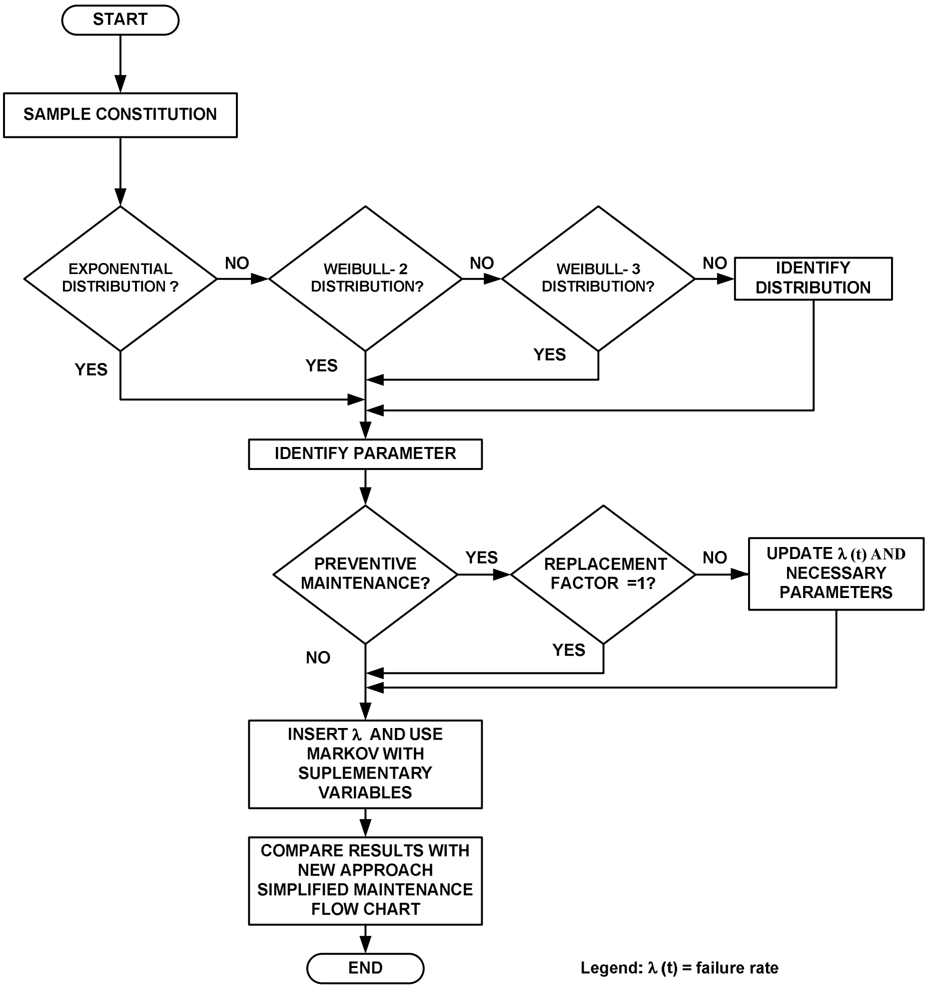

Figure 2 – Simplified Methodology

5. FAILURE DATA

The representative sample of the field data comes from the description of the Maintenance Service Orders. Starting with these data, a preliminary analysis is performed, for which engineering judgment is used to identify incomplete or undefined data. Next, the related data are compared to verify if there are any conflicting data, and if any complementary information need to be given for better explanation of the data. The collected data must be divided in accordance with the respective SSC, using the Pareto's method to select the failure processes to be analyzed.

From the Cumulative Distribution Function $[F(t)]$ of statistical distributions, whose linear regression models are shown in table 1, it is possible to find the correlation and determination coefficients, used for the least square method, that indicates the distributions candidates that initially better represent the failure data obtained. For the estimated $\hat{F}(t)$ that represents the fraction of observations below the *i*th observation, that is, the rank statistics, the choice was:

$$\hat{F}(t_i) = \frac{i-0{,}3}{n+0{,}4} \tag{1}$$

This rank is frequently used in statistics, as the median approach of the failures time plotting, as suggested by (Ebeling, 2005), since $\hat{F}(t)$ is skewed for values of "i" close to zero and close to "n". Φ^{-1} corresponds to the inverse standard normal distribution function.

Table 1 – Relationship between the "x" and "y" variables from linear regression and related statistical distribution

STATISTICAL DISTRIBUTION	x_i	y_i
exponential	t_i	$ln\{1/[1-F(t_i)]\}$
normal	t_i	$z_i = \Phi^{-1}[F(t_i)]$
lognormal	$ln\ t_i$	$z_i = \Phi^{-1}[F(t_i)]$
Weibull	$ln\ t_i$	$lnln\{1/[1-F(t_i)]\}$

Additionally, the maximum likelihood estimators (MLE) should be identified, if possible, for representing the failure data set, giving priority to the exponential and the two-parameter Weibull distributions. In the case of the exponential distribution, the MLE estimator for the failure rate for complete and censured data is given by

$$\hat{\lambda} = \frac{r}{\sum_{i=1}^{r} t_i + (n-r)t_r} = \frac{r}{T} \tag{2}$$

where r corresponds to the number of failures and time T is defined in Table 2. The t_r from censored type II data must be replaced by t^* for samples with censored type I data.

Table 2 – Calculation of total time test

DATA	T
Complete	$\sum_{i=1}^{n} t_i; \qquad r = n$
Type I censor	$\sum_{i=1}^{n} t_i + (n-r)t_*$
Type II censor	$\sum_{i=1}^{n} t_i + (n-r)t_r$

$\boldsymbol{t_i}$ = failure time; $\boldsymbol{t^*}$ = test time (Type I testing); ***tr*** = time of *r*th (Type II testing); ***n*** = total number of units at risk; e ***r*** = number of failures.

For the case of the Weibull distribution, the equation for estimating the MLE of the shape parameter β, for complete or censored data is given by:

$$g(\hat{\beta}) = \frac{\sum_{i=1}^{r} t_i^{\hat{\beta}} \ln t_i + (n-r) t_s^{\hat{\beta}} \ln t_s}{\sum_{i=1}^{r} t_i^{\hat{\beta}} + (n-r)\, t_s^{\hat{\beta}}} - \frac{1}{\hat{\beta}} - (1/r) \sum_{i=1}^{r} \ln t_i = 0 \tag{3}$$

where the estimated MLE for the characteristic life θ can be obtained from:

$$\hat{\theta} = \left\{ \frac{1}{r} \left[\sum_{i=1}^{r} t_i^{\hat{\beta}} + (n-r) t_s^{\hat{\beta}} \right] \right\}^{1/\beta} \tag{4}$$

where $\boldsymbol{t_s}$ is equal to 1, and $\boldsymbol{t_*}$ and $\boldsymbol{t_r}$, for complete data, are times of type I and type II censoring, respectively. The numerical solution, can be found by the Newton-Raphson method where

$$\hat{\beta}_{j+1} = \hat{\beta}_j - \frac{g(\hat{\beta}_j)}{g'(\hat{\beta}_j)} \tag{5}$$

and

$$g'(x) = \frac{dg(x)}{dx} \tag{6}$$

After choosing the theoretical distribution, a statistical test for goodness of fit of the failure data must be performed. In the case of the exponential distribution, the Bartlett's test can be used for complete and singly censored data, being recommended for samples of at least 20 failure data, Ebeling (2005).

The hypotheses are:

H_0: Failure times follow an exponential distribution; and

H_1: Failure times do not follow an exponential distribution.

The test statistics is given by

$$B = \frac{2r\{\ln[(1/r)\sum_{i=1}^{r} t_i] - [(1/r)\sum_{i=1}^{r} \ln t_i]\}}{1 + [(r+1)/6r]} \tag{7}$$

where t_i represents the time of failure of the *i*-th unit and r corresponds to the number of failures. The Bartlett's test uses the qui-square distribution table where the null hypothesis is accepted if:

$$\chi^2_{1-\alpha/2,r-1} < B < \chi^2_{\alpha/2,r-1} \tag{8}$$

and

$$Pr\{\chi^2 < \chi^2_{1-\alpha/2,r-1}\} = Pr\{\chi^2 < \chi^2_{\alpha/2,r-1}\} = \frac{\alpha}{2} \tag{9}$$

In the case of the Weibull failure distribution, the test of Mann, specifically developed by Mann, Schafer and Singpurwalla, uses the following hypotheses (Ebeling, 2005):

H_0: the failure times follow a Weibull distribution; and

H_1: the failure times do not follow a Weibull distribution.

This test can be summarized as follows:

$$k_1 = \left[\frac{r}{2}\right] \tag{10}$$

$$k_2 = \left[\frac{r-1}{2}\right] \tag{11}$$

$$Z_i = ln\left[-ln\left(1 - \frac{i - 0{,}5}{n + 0{,}25}\right)\right] \tag{12}$$

$$M_i = Z_{i+1} - Z_i \tag{13}$$

$$M = \frac{k_1 \sum_{i=k+1}^{r-1}[(lnt_{i+1} - lnt_i)/M_i]}{k_2 \sum_{i=1}^{k_1}[(lnt_{i+1} - lnt_i)/i]} \tag{14}$$

where k_1 and k_2 represent the integer portion of the result of the corresponding ratio and the failures data must be rank-ordered. The test consists on the comparison of the value of M from equation (14) with a determined critical value (F_{crit}). The condition $M < F_{crit}$ must be satisfied for the null hypothesis not to be rejected.

The value of F_{crit} will be obtained through the inverse function of the F distribution, where the number of degrees of freedom corresponds to $2k_2$ and $2k_1$, for the numerator and denominator, respectively. The value of the probability to be used for the data will correspond to the confidence interval (1 - α), where α is the level of significance.

6. RESTRICTIONS OF THE MARKOVIAN MODEL AND THE METHOD OF SUPPLEMENTARY VARIABLES

The Markov model considers the system states and their possible transitions, assuming that the transition rates between the states are constant, corresponding to the exponential distribution, where the probability transitions are only determined by the present state, without considering the previous ones. The basic Markov model has the advantage of being simple to be formulated and it is largely employed for practical cases, but it has the disadvantage of not considering aging processes, being necessary the use of a model that eliminates this restriction. This gap can be filled by the method of supplementary variables, based on the insertion of additional variables to cast the so called Non-markovian model into a Markovian one, Cox & Miller (1965).

7. DEVELOPMENT OF THE CASE STUDY

In order to analyze the adequacy of the proposed methodology, a case study was developed concerning the Auxiliary Feed Water System (AFWS) of Angra-1 Nuclear Power Plant. The reason for choosing this system is that it has few components and it is important for the plant safety, as long as it is related to the following critical functions: heat sink, sub-criticality, core cooling, integrity, containment and inventory (Araújo, 1998).

The AFWS consists of two trains, each with 100% capacity of supplying the design flow (to supply the steam generators, when safety related). One of these trains is composed by two motor-driven horizontal centrifugal pumps (AF-1A and AF-1B) with a 260 GPM capacity each, while the other one is composed by a turbine-driven pump (AF-2). Besides the pumps, the AFWS is composed by an Auxiliary Storage Water Tank (ASWT) common to both trains and controls, instrumentation, valves and pipeline associated with each train. The ASWT function is to send water to the steam generators, within one minute limit, after a signal generation from the reactor protection system (RPS) (Araújo, 1998).

With the plant in the normal operation mode, the system remains in a standby condition, automatically operating when demanded by the RPS.

The motor and the turbine pumps must be tested at least monthly, and a flow test must be performed for the steam generators, at least once in 18 months or during each refueling cycle, whichever comes first (Araújo, 1998).

The Technical Specifications (FURNAS, 1994) establish the following performance criteria for the AFWS:

The reactor shall not be made critical and the primary system temperature must not be higher than 177º C (350º F) unless the following requirements, among others, are satisfied:

- a minimum of 120.000 gallons (454,200 liters) of water must be available in the Auxiliary Storage Water tank (ASWT); and
- the turbine-driven pump and at least one of the motor-driven pumps, with valves and associated pipelines, must be operational.

If any of the requirements cannot be satisfied within 48 hours, with the plant in the operation or critic condition, then the plant must be placed in a cold shutdown condition. (Araújo, 1998).

Table 3 – AFWS Failures Data

RUNNING TIME UNTIL AWSS PUMPS FAILURE (IN HOURS)		
AF – 1A	**AF – 1B**	**AF – 2**
144	12	12
264	24	72
348	60	108
924	84	132
1584	300	144
	456	228
	480	240
	624	240
	1020	276
		360
		423
		468
		600

With the failure times given in Table 3 and using the linear regression models presented in Table 1, the failure rates of Table 4 are found.

Table 4 – AFWS Failures Rate

AFWS PUMPS FAILURE RATE (IN HOURS)		
AF – 1A	**AF – 1B**	**AF – 2**
Exponential Distribution	Exponential Distribution	Weibull Distribution
Failure Rate: $1.238 \times 10^{-3} h^{-1}$	Failure Rate: $2.363 \times 10^{-3} h^{-1}$	Shape Parameter (β): 1.074
Mean Time to Repair (MTTR) 6.5 h (*)	Mean Time to Repair (MTTR): 6.8 h (*)	Scale Parameter (θ): 302.2844 horas
		Mean Time to Repair (MTTR): 4.9 h (*)

The MTTRs were obtained from Pinho (2000).

Using simplified nomenclature of A, B and T for pumps AF-1A, AF-1B and AF-2, respectively, the following system states are obtained:

- State 1 – The three pumps are working $- ABT$;
- State 2 – Motor-driven pump A fails and the other two pumps are working – $\bar{A}BT$;
- State 3 – Motor-driven pump B fails and the other two pumps are working – $A\bar{B}T$;
- State 4 – Turbine-driven pump B fails and the other two pumps are working – $AB\bar{T}$;
- State 5 – Both motor-driven pumps are failed and the turbine-driven pump is working – $\bar{A}\bar{B}T$; and
- State 6 – System Total Failure – The system cannot supply the design flow - $\bar{A}\bar{B}\bar{T}$.

$P_4(t), P_5(t)\ e\ P_6(t)$ are considered as failure probabilities for our case study, since as established in FURNAS (1994) the turbine-driven pump and at least one motor-driven pump, with valves and associated pipelines, must be operational being the plant placed in a cold shutdown condition within 48 hours otherwise. This means that these three system states must be rigorously monitored.

The system state diagram is shown in Figure 3.

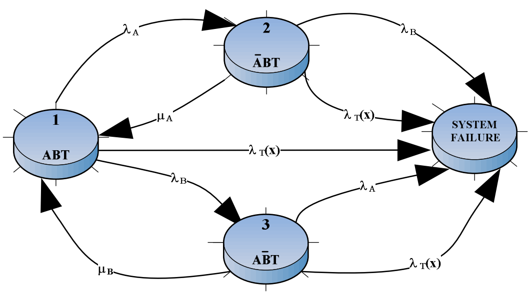

Figure 3 – States Diagrams to Auxiliary Feeding Water System

Using the following equations [whose details may be found in Pinho (2000)]

$$\frac{\partial p_1}{\partial x} + \frac{\partial p_1}{\partial t} = -[\lambda_A + \lambda_B + \lambda_T(x)]p_1(x,t) \tag{15}$$

$$\frac{\partial p_2}{\partial x} + \frac{\partial p_2}{\partial t} = -[\lambda_B + \lambda_T(x)]p_2(x,t) \tag{16}$$

$$\frac{\partial p_3}{\partial x} + \frac{\partial p_3}{\partial t} = -[\lambda_A + \lambda_T(x)]p_3(x,t) \tag{17}$$

$$\frac{dP_4}{dt} = \int_0^\infty \lambda_T(x)p_1(x,t)dx + \int_0^\infty [\lambda_B + \lambda_T(x)]p_2(x,t)dx + \int_0^\infty [\lambda_A + \lambda_T(x)]p_3(x,t)dx \tag{18}$$

with the following initial and boundary conditions

$$p_1(0,t) = \mu_A \int_0^\infty p_2(x,t)dx + \mu_B \int_0^\infty p_3(x,t)dx \tag{19}$$

$$p_2(0,t) = \lambda_A \int_0^\infty p_1(x,t)dx \tag{20}$$

$$p_3(0,t) = \lambda_B \int_0^\infty p_1(x,t)dx \tag{21}$$

$$p_1\ (0) = 1; \quad p_2\ (0) = 0; \quad p_3\ (0) = 0; \quad P_4\ (0) = 0;$$

$$P_i(t) = \int_0^x p_i(x',t)\,dx' \ , i = 1,2,3. \tag{22}$$

the system reliability figure of interest may be found.

Solving these equations by finite difference methods, considering a time period of 24h (which is longer than the required mission time for the AFWS), one finds that the system reliability is equal to 97.46%.

The next step, as displayed in Figure 2 is to compare the system reliability just found with the SSC performance monitoring as established by the expert panel. However, as the Maintenance Rule is not as yet implemented in Brazilian nuclear power plants and the reference parameter is not as yet defined, the possible outcomes concerning the AFWS are next discussed.

- Outcome # 1 – The reliability found is compatible with the performed monitoring and both the performance parameter and the reliability itself consider that the SSC functioning is appropriate. In this case the established monitoring policy should be kept;
- Outcome # 2 – The reliability found is compatible with the performed monitoring and both the performance parameter and the reliability itself consider that the SSC functioning is inappropriate. In this case, an intervention in the SSC is necessary;
- Outcome # 3 – The reliability found and the performed monitoring are not compatible, so that the performance parameter considers the SSC functioning adequate, while the reliability criterion does not. The methodology suggests that the discrepancy reason should be identified, by starting the algorithm; if it does not need to be corrected, the performance parameter checking should proceed; and
- Outcome # 4 – The reliability found and the performed monitoring are not compatible, so that the performance parameter considers that the SSC functioning is inadequate, while the reliability criterion considers it adequate. In this case, the methodology suggests the same procedure sequence as in the third outcome just discussed.

The importance of the algorithm as an indicator of the need of the process feedback is clear if one observes that while the occurrence of outcome # 3 or #4 triggers the feedback no matter what period has been established, the scheduled feedback in the Maintenance Rule is 24 months or after a plant refueling cycle

8. CONCLUSIONS

This paper presents an stochastic model for the reliability analysis of safety systems of nuclear power plants in the context of the plant qualified life extension, where aging mechanisms are of utmost importance and also taking into account the newly Maintenance Rule philosophy to be implemented in Brazilian plants.

The need for modeling aging mechanisms means that methods like the supplementary variables methods should be used, as discussed in Cox and Miller (1965).

On the other hand, new features should be considered when the Maintenance Rule is adapted to consider aging. This is of particular importance if one considers that the Angra 1 Nuclear Power Plants is under a qualified life extension process and its Maintenance Rule will need to consider this feature.

As shown in this paper, the extension of the Maintenance Rule for considering aging effects is quite simple and a personal computer can easily handle all calculations.

A natural extension of the work performed here is the more detailed consideration of safety systems. For this purpose, the need of plant-specific failure data is a demand.

The stochastic model here proposed assumes that failure and repair times are independent and, in most cases, identically distributed. When this is not the case, other stochastic approaches should be considered. One that is being currently used for this purpose is the use of stochastic point processes, as discussed in Jacopino (2005).

9. REFERENCES

ARAUJO, Jefferson B. *Um Modelo de Manutenção Centrada na Confiabilidade Aplicada ao Sistema de Água de Alimentação Auxiliar de uma Usina Nuclear*. Dissertação de Mestrado, COPPE/UFRJ, Rio de Janeiro, RJ, 1998.

COX, D. R., MILLER, H. D. *The Theory of Stochastic Processes*, London. Chapman and Hall, 1965.

EBELING, C. E. *An Introduction to Reliability and Maintainability Engineering.* Estados Unidos da América. Waveland Press, Inc, 2005.

FURNAS, Furnas Centrais Elétricas S. A. *Final Safety Report*. Central Nuclear Almirante Álvaro Alberto, Angra I, Westinghouse Eletric Corporation, FSAR, Rev nº 24, Brasil, 1994.

JACOPINO, A. G., Generalization and Bayesian solution of the general renewal process for modeling the reliability effect of imperfect inspection and maintenance based imprecise data, Ph.D. thesis, Department of Mechanical Engineering, University of Maryland, 2005.

NEI, Nuclear Energy Institute. *Industry Guideline for the Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.* NUMARC 93-01 3rd Revision, Washington, Nuclear Management and Resource Council, Inc., 2000.

NRC, Nuclear Regulatory Commission. *Lessons Learned From Maintenance Rule Baseline Inspections.* NUREG 1648, Washington, 1999.

NRC, site http://www.nrc.gov/reactors/operating/ops-experience/maintenance-effectiveness/mainten-effect-files/simp-maint-rule.gif, 2009.

PINHO, M. O. *Sobre a Aplicação de Sistemas de Equações Diferenciais Parciais e Ordinárias de Primeira Ordem à Confiabilidade de Sistemas de Segurança sob Envelhecimento*. Tese de Doutorado, COPPE/UFRJ, Rio de Janeiro, RJ, 2000.

10. RESPONSIBILITY NOTICE

The authors are the only responsible for the printed material included in this paper.