METHODOLOGY TO ANALYSIS OF AGING PROCESSES OF CONTAINMENT SPRAY SYSTEM

Diogo da Silva Borges [1a], Deise Diana Lava [1b], Maria de Lourdes Moreira [1c], Antonio Cesar Ferreira Guimarães [1d] and Lucinda Fernandes da Silva[1e]

[1] Instituto de Engenharia Nuclear (IEN / CNEN - RJ)
Rua Hélio de Almeida 75
21941-906 Rio de Janeiro, RJ - Brasil - Caixa-Postal:68550
[a] diogosb@outlook.com, [b] deise_dy@hotmail.com , [c] malu@ien.gov.br, [d] tony@ien.gov.br, [e] lucinda@ien.gov.br

ABSTRACT

This paper presents a contribution to the study of aging process of components in commercial plants of Pressurized Water Reactors (PWRs). The motivation for write this work emerged from the current perspective nuclear. Numerous nuclear power plants worldwide have an advanced operating time. This problem requires a process to ensure the confiability of the operative systems of these plants, because of this, it is necessary a methodologies capable of estimate the failure probability of the components and systems. In addition to the safety factors involved, such methodologies can to be used to search ways to ensure the extension of the life cycle of nuclear plants, which inevitably will pass by the decommissioning process after the operating time of 40 years. This process negatively affects the power generation, besides demanding an enormous investment for such. Thus, this paper aims to present modeling techniques and sensitivity analysis, which together can generate an estimate of how components, which are more sensitive to the aging process, will behave during the normal operation cycle of a nuclear power plant.

1. INTRODUCTION

The term aging for nuclear power plant is defined by the NRC (Nuclear Regulatory Commission) as the cumulative time-dependent systems degeneration, structures or components in a nuclear plant that if not investigated, may compromise the safety and operation of an installation [1]. The study directed to the aging process is due to the need to analyze the effects caused by exponential increase of failure probability, defined as the number of times (X) that an event (E) in a sample space (S) can be repeated from a number (n) of repeated experiments whose results are shown by the sample space (S) [2] according to the equation (1), or function loss of a given component that inevitably always will be subject to degenerative factors over its operational life, as oxidation processes, corrosion, manufacturing defects, and even by possible fission products. The nuclear power plants aging process is handled in its manufacturing process and mitigated through guarantees that everything will work with sufficient margins to ensure the minimum required operating time.

$$P(E) = \lim_{n \to \infty}(X/n) \quad (1)$$

This concept directs vision of reliability and safety for non-destructive testing methods, like the Fault Tree Method, for materials more susceptible to degenerations, such as pumps and valves, whose objective is to search to understand the aging process effects and to ensure the detection of early signals of its interaction with a system during its operating time. With a research program about nuclear power plants aging process (known as NPAR-Nuclear Power Aging Research [1], in the case of USA units), besides ensuring safety, it is possible extend life an installation, thus obtaining, their license renewal. Although various aging techniques presenting inaccuracies they are heavily used to obtain a good safety margin.

In this paper will be present a simulation about the aging process in the Containment Spray Injection System (CSIS) of a Pressurized Water Reactor (PWR) using the Fault Tree (FT) Method, defined as "[…] a graphic model of various parallel and sequential combinations of fault that will result in the occurrence of a predefined undesired event." [3]. The FT has capacity to present the logic of events that lead to system unavailability, capture frequency estimation of events, to model and calculate hazardous events frequency (before its happening) and help develop and evaluate protective layers and analysis of sensitivity and importance with the use of technics like Monte Carlo Method, Birnbaum Importance Measure, Fussell-Vesely Importance Measure, Risk Reduction Ratio and Risk Increase Ratio. The Monte Carlo Method and Fussell-Vesely Importance cited are used in this paper to determine the system unavailability probability and the most sensitive events to the aging process, in other words, to seek components that show significant an increase in his failure probability rate during its operational life in a nuclear power plant and to determine the system unavailability.

2. METHODOLOGIES

2.1. Fault Tree Method structure

The Fault Tree Method uses probabilistic combinations of basic events, individual failures, in order to trace the events that contribute directly to the main event (Top Event) [3]. The characterization is made using combinations of logical operators, which in turn indicate mathematical operations under which relate basic events (Fig. 1). The combination between basic events is made in order to integrate them as subsets of a universe, in which the sets where the analysis is made can to contribute to the system failure or a possible vital component for the installation, thus, the Fault Tree is characterized as a failure method, in other words, this method search individual failures (basic events) that lead to a possible collapse of structures.

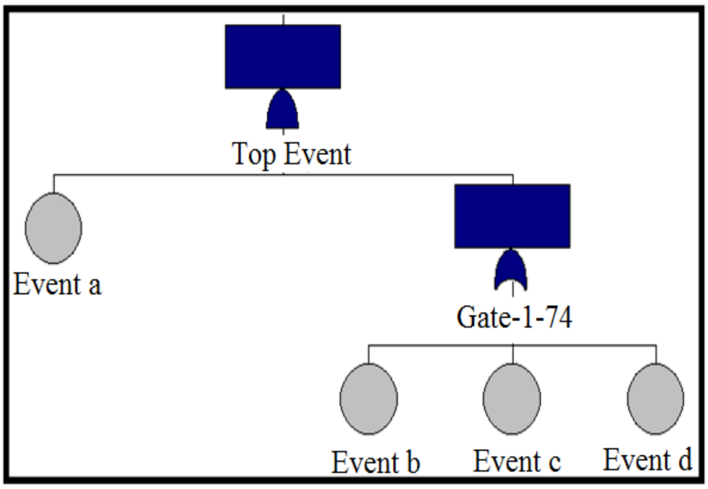

Figure 1. Fault Tree Model.

Another important part of method is use of minimal cut sets, minimal faults combination that result in occurrence of an interest event [4]. Due to the necessity for a high level of data processing, the method needs to somehow to use algorithms that can “scan” its data, identify and select “a calculation path” shorter in order to make the most effective method during to calculation of the system unavailability. The necessity to seek simplifications, which in turn leads to optimization method, requires an approach to recursive algorithms, loading and restructuring, top gate determination, loop error detection, gates conversion complemented, event house suppression, modules versus independent sub trees, creation and modules determination, independent events determination, gates and sub trees determinations, determination of gates levels, fault trees reductions, minimal cut sets truncation, hiding intermediate results, absorption of minimal cut sets and fault trees gates expansions.

The Fault Tree Method usually approach complex systems that often require the use of a computational language for data processing. Applying the concept of minimal cut sets [2] it is possible to combine different failure path analysis in order to obtain a single expression. Through utilization of sets laws is possible to obtain an expression that can be processed more efficiently.

2.2. Monte Carlo Method

This technique allows the use of random numbers and mathematical statistical models to simulate real systems [5], in other words, through this technic is possible to calculate multiple scenarios modeled by repetitions of values of probability distributions for uncertainty variables. His study is only possible through the effectuation of several simulations or iterations to estimate a central tendency of numerical variations, which makes it able to solve deterministic problems more complicated.

Monte Carlo randomly generates uncertainty variable values to simulate a model, such that, for each uncertainty variable the values are assigned with the probability distribution. Distribution types can include regular, triangular, uniform, lognormal, Bernoulli, binomial and Poisson distributions.

In most cases the simulation done by this technique uses the Weibull equation (as well as the specific condition (β = 1) to the exponential distribution) [5], to be relatively simple and describe the weakest links of the failure mechanisms [6]. The Weibull equation used in Monte Carlo Method is resolved by the time constant (t), whose relationship between Weibull and cumulative distribution function (c.d.f), F (t), t and β is given by equation (2).

$$t = \mu . ln[(1/(1 - F(t))]^{1/\beta} \tag{2}$$

Such that:

- t is the operating time variable;
- β is the Weibull Distribution shape parameter;
- μ is the Weibull Distribution scale parameter.

2.3. Fussell-Vesely Importance Measure

Fussell-Vesely value is an importance measure directed to the cut sets. According to this measure, the importance of a component depends on the order and number in which the cut sets appears. The Fussell-Vesely Importance Measure indicates a fractional reduction in risk associated with a decrease in frequency of events (E_i), such that:

$$I_{FV} = \frac{P(TE|E_i = \langle E_i \rangle) - P(TE|E_i = 0)}{P(TE|E_i = \langle E_i \rangle)} \tag{3}$$

Where, $P(TE|E_i = \langle E_i \rangle)$ and $P(TE|E_i = 0)$ is given as minimal cut sets upper bound or evaluating minimum failure rate of a Fault Tree with the basic events set equal to its average value and zero, respectively.

2.4. Sensitivity and Importance Analysis

Techniques like Monte Carlo (MC), Birnbaum and Fussell-Vesely Importance Measure allow knowing how it behaves the stochastic uncertainty inheritance, characterized as the probabilistic data accumulation that constitute a part or imperfections class in the information that try to model a real system behavior [5]. Through these analyzes it is possible to ensure performance to estimating the impact value due to changes in Fault Tree structure, or even, indicate the sensitivity fraction or frequency in relation to basic events present in a cut set. Other techniques of sensitivity and importance analysis can be cited, like Risk Reduction Ratio (RRR) and Risk Increase Ratio (RIR).

The Monte Carlo Method and Fussell-Vesely Importance Measure described in this sub item are applied in the real case, Containment Spray Injection System evaluation, in order to calculate the probability failure of the top event and determine the importance and contribution of each cut set to the system unavailability.

3. METHODOLOGY APPLICATION

The methodology application is directed to Containment Spray Injection System of a PWR Nuclear Power Plant (see WASH – 1400 [7]), system responsible by the containment cooling. The Spray System together with the ventilation system (Fan Cooler System) comprise a safety system third and last barrier level. Besides being responsible for containment temperature control, these systems are connected to the pressure control.

The Containment Spray System is the third safety level responsible for containment structural safety, which is the last barrier against leakage of radioactive products in accident case. The Spray Containment System offers a cold water mixture with boron through sprays from Refueling Water Storage Tank (RWST) [8]. The main function of this system is pressure drop through the temperature control. The Injection System consists of two redundant subsystems of RWST connected to containment. The recirculation phase was not considered here.

The Containment Spray Injection System is composed of two identically equal systems capable to supply 3,200 GPM from RWST to atmosphere through spray heads arranged in 360º. Each head contains 368 nozzles spaced, located 120 feet above of containment base. Both subsystems can, in an emergency, to use 350,000 gallons from RWST [7]. The reservation is arranged such that can guarantee that sodium hydroxide to be preferentially extracted by the Spray Injection System, since the hydroxide is preferably used in the administration of containment volume for initial removal of fission products. In the Fig.2 it is possible to see a simple scheme that shows the components distribution of the CSIS.

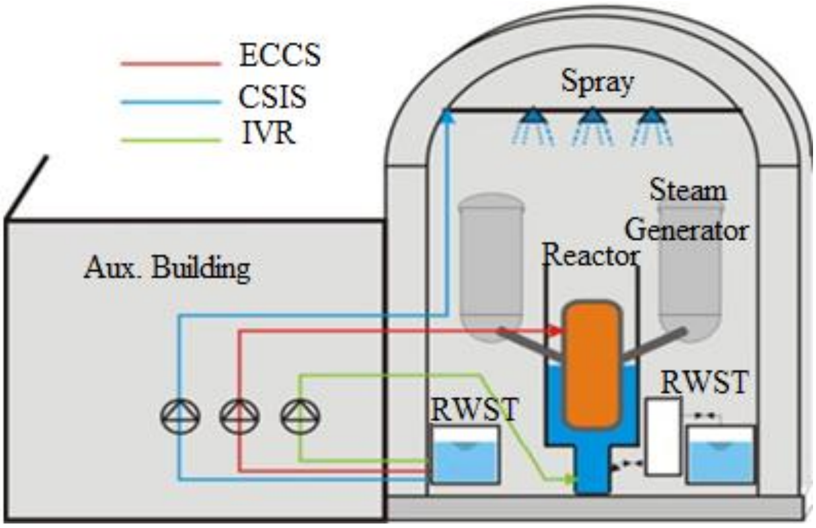

Figure 2. PWR cooling systems [9].

The Fig.3 shows a simplified diagram of the Containment Spray Injection System. The valves positions shown in this figure are given of form to represent the normal plant operation. For the operation of two spray subsystems, the valves V5 or V6 and V8 or V7 must be opened and the pumps P1 and P2 should enter into operation state. The Valves V1 and V3 should receive a signal from the Consequence Limiting Control System (CLCS) to ensure that other valves are closed during the operation of the Containment Spray System or open them, if they have been incorrectly closed [7].

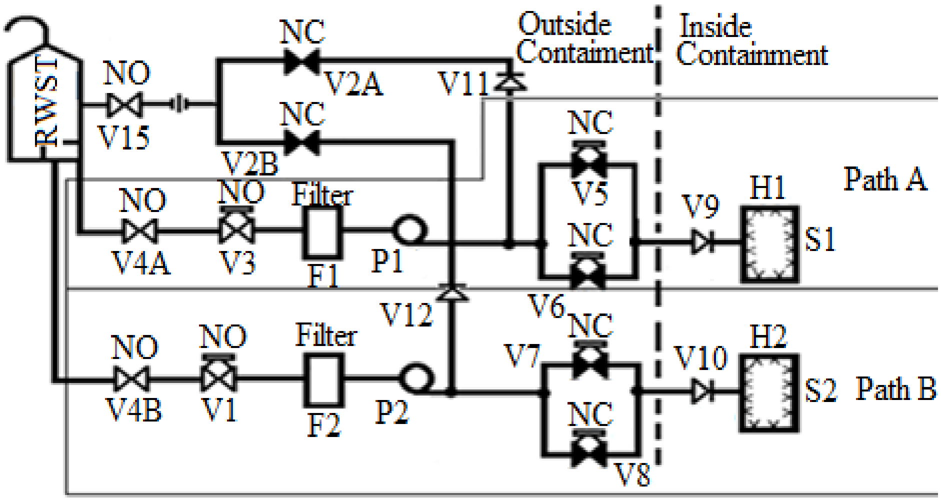

Figure 3. Containment Spray Injection System Simplified Flow Diagram [4].

3.1. Containment Spray Injection System Fault Tree

The Injection System Fault Tree consists of a main tree (Fig.4), linked to another three sub trees (Fig. 5). The main tree is composed of thirty-five basic events, statistically independent, being that probability of the top event of teen sub trees are calculated and converted in basic events, five gates and one top event.

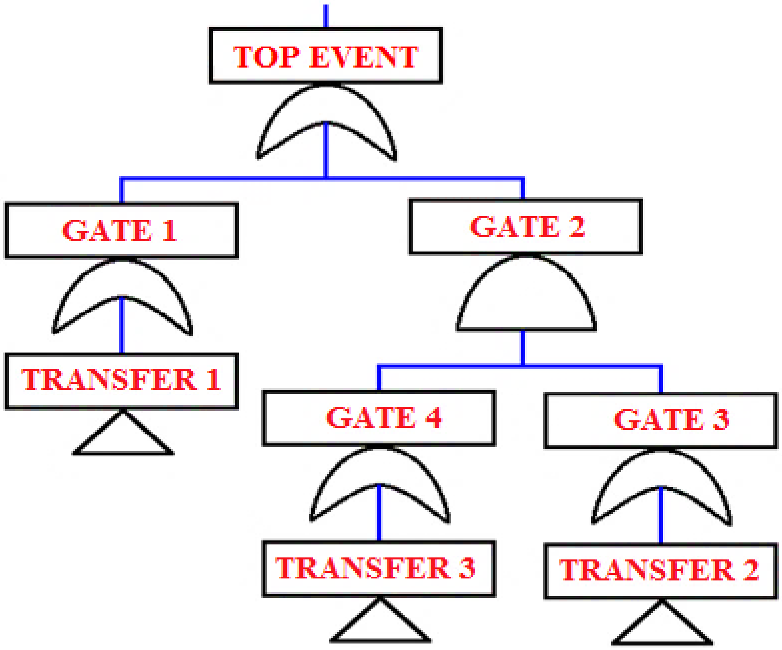

Figure 4. Main Fault Tree of CSIS [7].

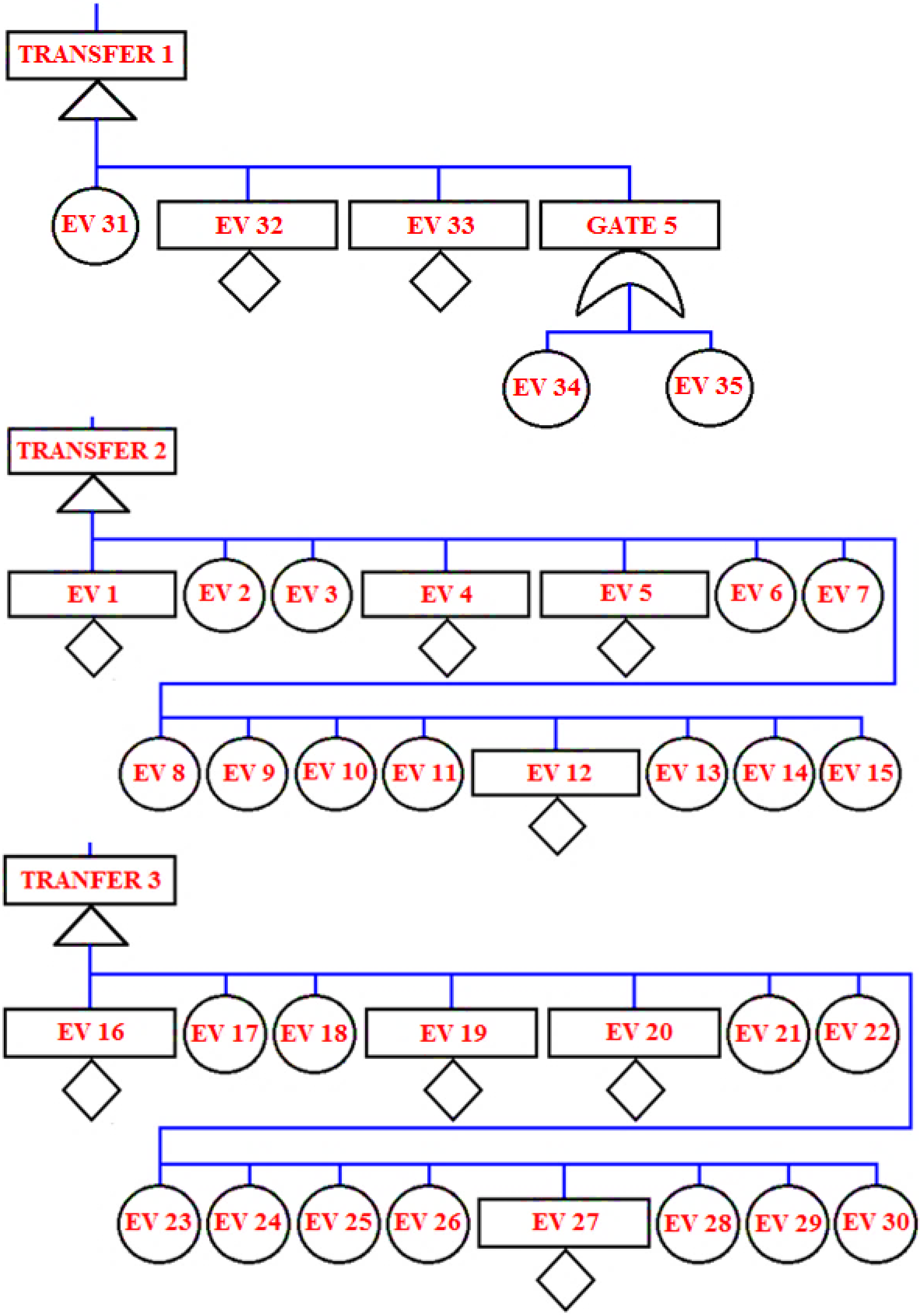

Figure 5. Sub Trees of CSIS [7].

The failure probabilities and description of basic events from the Fig.5 are given in the Tab.1 (see WASH – 1400 [7]). The flagged events (Ev*) indicate that are results from sub trees.

Table 1. Events and Failure Probabilities.

EVENT	DESCRIPTION	FAILURE PROBABILITY
Ev 1	Operator error manual valve (V4A) is left closed	1.000E-003
Ev 2	Failures in CONT circuitry closes MOV (System A)	1.000E-999
Ev 3	MOV (V1) inadvertently closes	1.000E-004
Ev 4	Operator error manual valve (V2A) is left open	1.000E-002
Ev 5	Line filter (F1) in A suction is plugged	1.100E-004
Ev 6	Motor drive of pump (P1) clutch disengages	3.000E-004
Ev 7	ELEC motor from pump (P1) fails to deliver SUFF Torque	1.000E-999
Ev 8	Failures in ELEC controls cause Pump (P1) not start	1.000E-003
Ev 9	Pump (P1) Fails to Start	1.000E-003
Ev 10	Pump (P1) Discontinues Running	1.500E-005
Ev 11	Check Valve (V9) Fails Closed	1.000E-004
Ev 12	Spray System (S1) Nozzles	1.300E-004
Ev 13*	Circuit fails to command pumps (P1) and valves (System A)	4.600E-003
Ev 14*	No 480 V power available to pump (P1) motor circuit breaker	4.100E-005
Ev 15*	No 125 V available to pump (P1) control circuit	1.100E-006
Ev 16	Operator error manual valve (V4B) is left closed	1.000E-003
Ev 17	Failures in CONT circuitry closes MOV (System B)	1.000E-999
Ev 18	MOV (V3) inadvertently closes	1.000E-004
Ev 19	Operator error manual valve (V2B) is left open	1.000E-002
Ev 20	Line filter (F2) in A suction is plugged	1.100E-004
Ev 21	Motor drive of pump (P2) clutch disengages	3.000E-004
Ev 22	ELEC motor from pump (P2) fails to deliver SUFF Torque	1.000E-999
Ev 23	Failures in ELEC controls cause Pump (P2) not start	1.000E-003
Ev 24	Pump (P2) Fails to Start	1.000E-003
Ev 25	Pump (P2) Discontinues Running	1.500E-005
Ev 26	Check Valve (V10) Fails Closed	1.000E-004
Ev 27	Spray System (S2) Nozzles	1.300E-004
Ev 28*	Circuit fails to command pumps (P2) and valves (System B)	4.600E-003
Ev 29*	No 480 V power available to pump (P2) motor circuit breaker	4.100E-005
Ev 30*	No 125 V available to pump (P2) control circuit	1.100E-006
Ev 31	RWST rupture	1.000E-999
Ev 32	Undetected RWST leakage	1.000E-999
Ev 33	RWST 8´´ vent plugged	4.400E-007
Ev 34*	Power failures cause discharge valve to stay closed (A & B)	1.000E-005
Ev 35*	Power failures cause discharge valve to stay closed (A & B)	3.000E-004

3.2. Results and Discussions

The first stage about the aging study of the Containment Spray Injection System consists in the cut sets selection that have greater importance to system unavailability. The Fault Tree shown in the Fig.4 generates a hundred and seventy two cut sets. The table below (Tab.2) presents twelve of the most important cut sets for the failure process (89.87% of the system unavailability), with their respective frequencies, Fussell-Vesely Importance Measures and contributions (in percent) to system general failure. The cut sets were selected based on their contribution for the system failure.

Table 2. Cut Sets Participation in System Unavailability.

Nº CUT SET	FREQUENCIE	FUSSELL-VESELY	% TOTAL	EVENT
1	3.000E-004	4.624E-001	46.24	Ev35
2	1.000E-004	1.541E-001	15.41	Ev4, Ev19
3	4.600E-005	7.091E-002	7.09	Ev4, Ev28
4	4.600E-005	7.091E-002	7.09	Ev19, Ev13
5	2.116E-005	3.262E-002	3.26	Ev13, Ev28
6	1.000E-005	1.541E-002	1.54	Ev4, Ev16
7	1.000E-005	1.541E-002	1.54	Ev9, Ev19
8	1.000E-005	1.541E-002	1.54	Ev1, Ev19
9	1.000E-005	1.541E-002	1.54	Ev8, Ev19
10	1.000E-005	1.541E-002	1.54	Ev24, Ev4
11	1.000E-005	1.541E-002	1.54	Ev23, Ev4
12	1.000E-005	1.541E-002	1.54	Ev34

The second stage of aging analysis is done through attribution of multipliers (Tab.3) in the failure probability of events presents in the Tab.2 and calculations of the system unavailability. The multipliers are positives numbers incremented in the failure probability of the basic events. In the Tab.3 the “General” represent the system unavailability due to same probabilistic variation of all basic events, assuming that they get older equally.

Table 3. System Unavailability.

BASIC EVENT	FAILURE RATE	SYSTEM UNAVAILABILITY MULTIPLICATIVE FACTOR			
		1x (Monte Carlo)	2x	5x	10x
Ev1	1.000E-003	6.487E-004	6.671E-004	7.223E-004	8.142E-004
Ev4	1.000E-002	6.487E-004	8.326E-004	13.84E-004	23.03E-004
Ev8	1.000E-003	6.487E-004	6.671E-004	7.223E-004	8.142E-004
Ev9	1.000E-003	6.487E-004	6.671E-004	7.223E-004	8.142E-004
Ev13	4.600E-003	6.487E-004	7.333E-004	9.870E-004	14.10E-004
Ev16	1.000E-003	6.487E-004	6.671E-004	7.223E-004	8.142E-004
Ev19	1.000E-002	6.487E-004	8.326E-004	13.84E-004	23.03E-004
Ev23	1.000E-003	6.487E-004	6.671E-004	7.223E-004	8.142E-004
Ev24	1.000E-003	6.487E-004	6.671E-004	7.223E-004	8.142E-004

Continuation of the Table 3.

Ev28	4.600E-003	6.487E-004	7.333E-004	9.870E-004	14.10E-004
Ev34	1.000E-005	6.487E-004	6.587E-004	6.887E-004	7.387E-004
Ev35	3.000E-004	6.487E-004	9.486E-004	18.48E-004	33.48E-004
General	--------------	6.487E-004	19.15E-004	93.97E-004	338.5E-004

Through of calculations of the system unavailability variation due the probability increased of each basic event present in the cut set from the Tab.2, we then have a graph that show the system unavailability progression due the susceptibility of each of its component to the aging process. The Fig.6 shows the system unavailability graph due to the aging process.

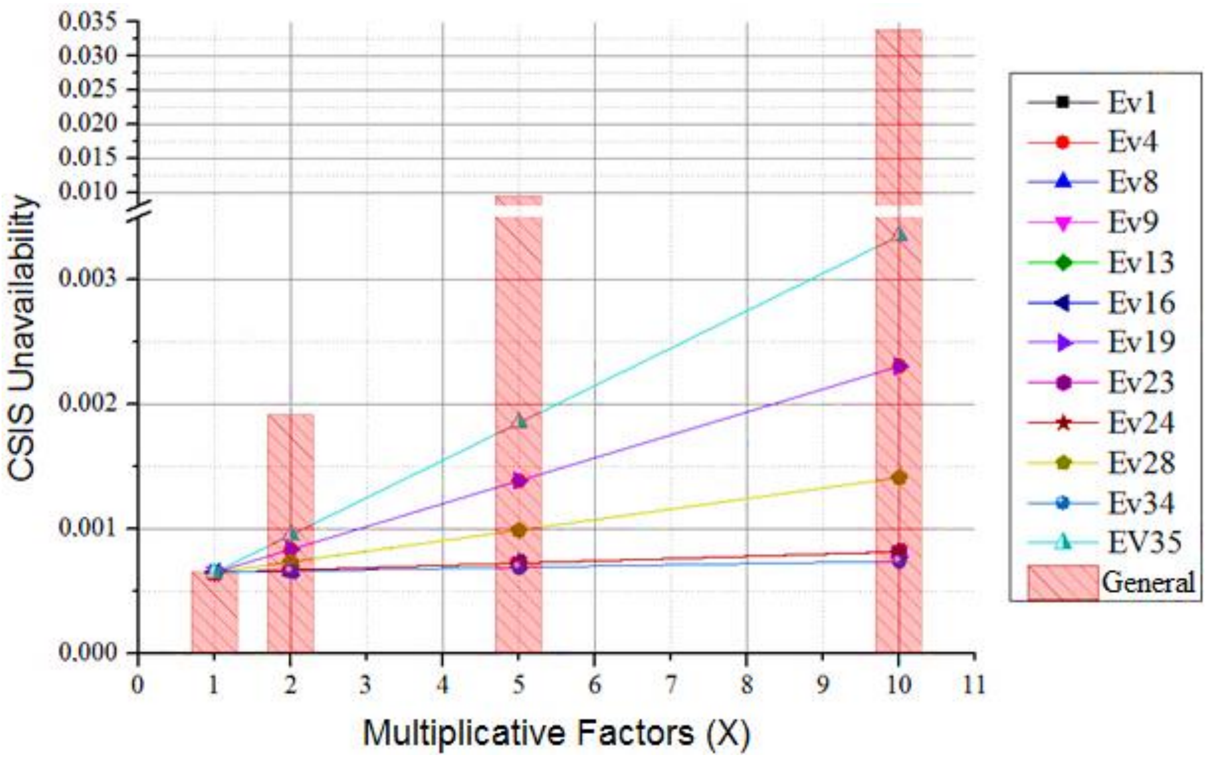

Figure 6. Behavior of CSIS Unavailability.

A simple way to analyze the effect of the aging on CSIS can be made by converting the failure probabilities presented in the Tab. 3 for the percentage form. In the Fig. 7 is shown a graph with the variation percentage (based on the multiplier 2x, 5x, 10x) of the probability of failure of the most sensitive components to the aging of the Containment Spray Injection System.

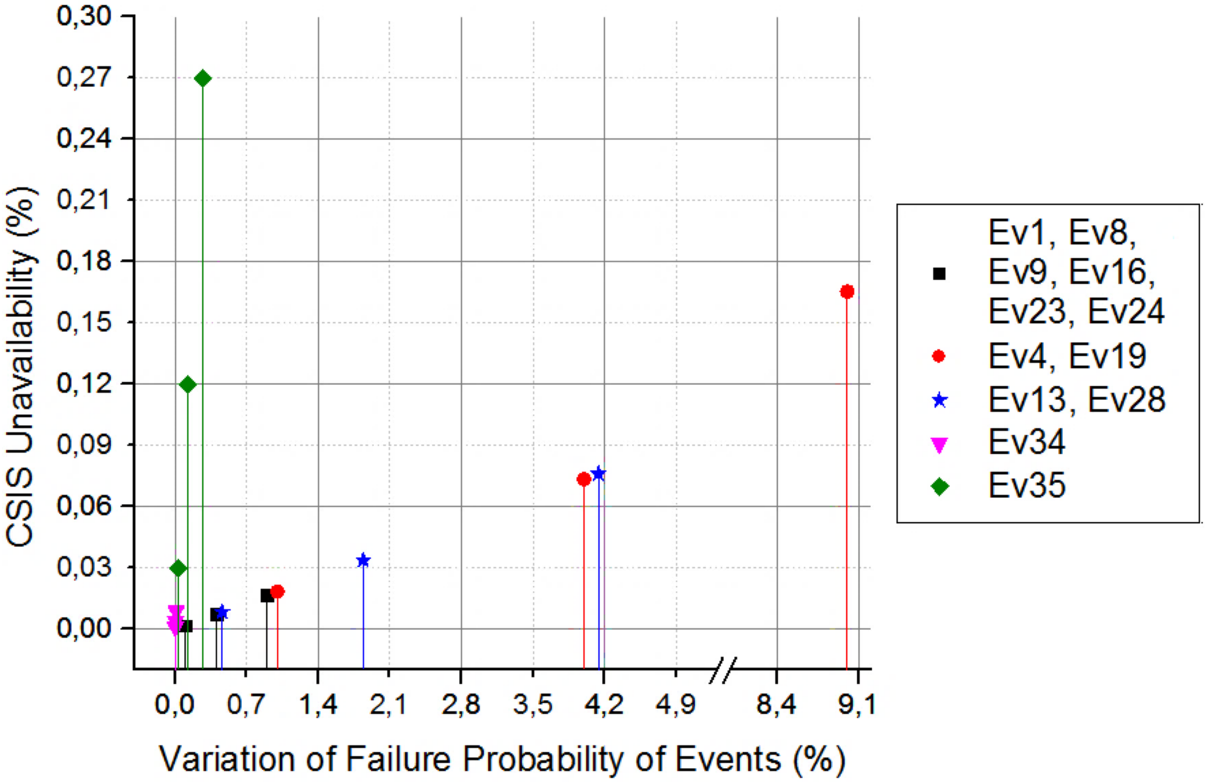

Figure 7. Progression Percentage of Unavailability of the CSIS.

3. CONCLUSIONS

The term aging for nuclear power is defined by the Nuclear Regulatory Commission (NRC) as the cumulative time-dependent systems degeneration, structures or components in a nuclear plant that if not investigated, may compromise the safety and operation of an installation. The greatest concern with aging is due to its capacity to modify vital properties of material structure in question.

The study about aging effect portrays the need to employ Probabilistic Safety Assessments (PSAs) related to progressive materials degeneration, in elapse of the operational life of a nuclear power plant. Besides the study to ensure safety, preventing that vital components come in fault state by operating time, the economic factor becomes another important issue to be analyzed, since prolongation of operation leads to considerable profits compared with decommissioning costs and building a new installation. In the case of Angra I and II Nuclear Power Plant would be interesting the development of a study similar to the NPAR.

According with results obtained in the graphs (Fig.6 and Fig.7) it is important to note the increase in the system unavailability due to contribution of components susceptible to the aging process. Through of changes in their properties, these components contribute directly in the increased of the system loss function probability, showing the importance of detailed studies about the choice of materials in regarding the need for workload and external factors that will be subjected to during its operational life. This study is able to ensure the safety of the system by identifying of components that need to be replaced into a period of time less than others, besides to enable explore all potential that the system can offer. Such identification impacts directly in control of life extension qualified, bringing benefits in relation to safety and economy of nuclear power plants.

It is important emphasize that other methodologies can be used for uncertainty analysis of a system, such as:

- Petri Net;
- Dynamic Fault Tree;
- Markov Cell-to-Cell;
- Bayesian Method;
- Black-Box Method.

These methods have the disadvantage of requiring a large amount of information to be applied, apart from requiring a considerable time for processing. Different of them, the Fault Tree Method requires a smaller amount of information to be applied and it is able to generate quick results with a good safety margin. Due to these characteristics, its application has been made safe, reliable and popularized in the nuclear environment, especially in the USA.

With the method described it was possible to obtain the function that governs how sensitive components CSIS influence the behavior of the system during a period of its operation. This determination allows the use of the time factor as an implicit variable of the analysis. The advantage of this determination is the possibility to measure the increased availability of the system through the characterizing the failure probability of a component by operating time, which can be obtained by non-destructive tests for example. Without the application of this

4. REFERENCES

1. U.S. Nuclear Regulatory Commission, “NUREG-1144 Rev. 1: Nuclear Plant Aging Research (NPAR) Program Plan”, Washington DC, pp. 1-135 (1987).
2. C. L. Jhon, J. M. Norman, *Risk and Safety Analysis of Nuclear Systems,* John Wiley & Sons, New Jersey, United States of America (2011).
3. U.S. Nuclear Regulatory Commission, "NUREG-0492: Fault Tree Handbook", Washington DC, pp.IV-1 - IV-2 (1981).
4. Z. Enrico, *An Introduction to the Basics of Reliability and Risky Analysis*, World Scientific Publishing, New Jersey, United States of America (2007).
5. F. S. Rudolph, *Handbook of Reliability*, *Availability*, *Maintainability and Safety in Engineering Design*, Springer, London, United Kingdom (2009).
6. W. M. Joe, *Reliability Physics and Engineering*, Springer, New York, United States of America (2012).
7. U.S. Nuclear Regulatory Commission, "WASH-1400: Reactor Safety Study", Washington DC, pp. II-111 - II-115 (1975).
8. P. Hoagan, *Safety and Risk Modeling and Its Applications*, Springer, London, United Kingdom (2011).
9. L.Kihwan, L. Kang-Heon, L. L Jeong, H. J. Young, L. Phill-Seung,"A new design concept for offshore nuclear power plants with enhanced safety features", *Nuclear Engineering Design*, United States of America, **Vol. 254**, pp. 10-11 (2013).