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Nondeterministic method to analysis of the aging effects in PWR power plants components

Diogo da Silva Borges *, Deise Diana Lava, Antonio Cesar Ferreira Guimarães, Maria de Lourdes Moreira

Instituto de Engenharia Nuclear (IEN/CNEN – RJ), Rua Hélio de Almeida 75, Caixa-Postal: 68550, 21941-906 Rio de Janeiro, RJ, Brazil

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ABSTRACT

This paper presents a contribution to the study of aging process of components in commercial plants of Pressurized Water Reactors (PWRs). The analysis is made through application of the Fault Trees Method, Monte Carlo Method and Fussell–Vesely Importance Measure. The approach of the study of aging in nuclear power plants, besides giving attention to the economic factors involved directly with the extent of their operational life, also provide significant data on security issues. In Brazil, for example, the Angra 1 Nuclear Power Plant invested $ 27 million to perform corrective actions in its network installation. This fact has generated an estimated operating life extension of Angra 1 in twenty years, offering great economy compared with building cost of a new plant and anterior decommissioning, if it had reached the time operating limit of forty years. The extension of the operating life of a nuclear power plant must be accompanied by a special attention to the components of the systems and their aging process. With the application of the methodology (aging analysis of the injection system of the containment spray) proposed in this paper, it can be seen the increase in the rate of component failure, due the aging process, generates the increase in the general unavailability of the system that containing these basic components. The final results obtained were as expected and may contribute to the maintenance policy, preventing premature aging process in Nuclear Plant Systems.

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 (U.S. Nuclear Regulatory Commission, 1987). The greatest concern with aging is due to its capacity to modify vital properties of material structure in question, such as:

- Expansion capacity;
- Mechanics capacity;
- Dielectric strength;
- Fatigue capacity;
- Ductility.

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) Jhon and Norman (2011) according to the Eq. (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:

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

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 (U.S. Nuclear Regulatory Commission, 1987), in the case of USA units), besides ensuring safety, it is

* Corresponding author.
E-mail addresses: diogosb@outlook.com (D.S. Borges), deise_dy@hotmail.com (D.D. Lava), tony@ien.gov.br (A.C.F. Guimarães), malu@ien.gov.br (M.L. Moreira).

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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.

The NPAR born as an engineering research program of NRC for Nuclear Power Plants aging study and monitoring of degeneration of components and systems. The first program discussion was given with the NUREG-1144 Rev. 1 document (U.S. Nuclear Regulatory Commission, 1987) in 1987 revised and worked to the course of years, as in 1990 with the NUREG-5507 document (U.S. Nuclear Regulatory Commission, 1990), in 1991 with the NUREG-1144 Rev. 2 document (U.S. Nuclear Regulatory Commission, 1991), 1993 with the OTA-E-575 document (U.S. Congress, Office of Technology and Assessment, 1993) in 2000 and 2003 with two license renewal cases examples in two U.S. nuclear plants with the NUREG-1723 (U.S. Nuclear Regulatory Commission, 2000) and NUREG-1782 documents (U.S. Nuclear Regulatory Commission, 2003). It was stated in the NPAR Program that these documents would be reviewed periodically. The revisions would reflect the experience acquired in program implementation and that would be incorporated comments received from the NRC industry codes, standards committees, national and foreign organizations and institutions, thus making this document essential for learning of aging effects through operational experience.

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." (U.S. Nuclear Regulatory Commission, 1981). 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 as technique to aging effects methodology

Using the Fault Tree concept is possible to characterize the failure probability of a system or equipment through probabilistic analysis of individual components present, whose data are obtained through operational experience. The technique used is directly related to probabilistic analysis of individual events that will trigger an events process, which in turn will result in a failure process. The understanding of calculations used by this technique for determining of the system unavailability is only possible with the study about probability concepts, as its laws and rules, and independent events.

The aging concept emerges of cut sets analysis, faults combination presents in a fault tree that result in occurrence of an interest event, where are done identifications of components susceptible to aging effects with the use of Fussell–Vesely Importance Measure, variations of failure probability with the use of multipliers and calculations of the system unavailability with the Monte Carlo Method.

2.1.1. 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) U.S. Nuclear Regulatory Commission, 1981. The characterization is made using combinations of logical operators, which in turn indicate mathematical operations under which relate basic events. 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.

Logical operators can be defined as logical connection functions (Hoagan, 2011), mathematical operations between the sets to be analyzed, either by literal or graphical presentations. Three operators, AND operator ($\cap$), OR ($\cup$) David, 2004 and NOT (/), are used in this paper. These operators are related to basic operations between sets, as laws absorption, complementation and switching for example, which enables optimizations of calculations pertinent to method through the determination of minimal cut sets.

Another important part of method is use of minimal cut sets, minimal faults combination that result in occurrence of an interest event (Enrico, 2007). 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.

2.1.2. Graphical and literal Fault Tree presentation

The Fault Tree graphical presentation is done to organize the events with lines and gates (symbols that represent logical operators). This gates can be found in seventeen different types, thus representing, specific roles during the calculations. The graph should contain only one Top Event, represented by a gate located on top, such that a single line will connect it to basic events and gates. The following a simple example (Fig. 1) of Fault Tree.

Another way to present a tree is through of letters and symbols of corresponding logical operators. It is possible to make a transcript of the graphical form to literal and literal to graphical just

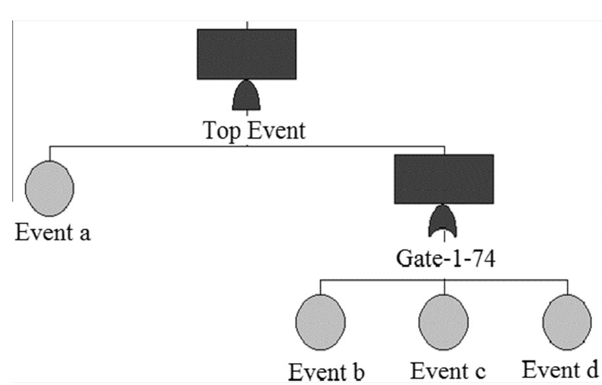

Fig. 1. Fault Tree Model.

watching the basic events interaction with gates (operators). For example, the model from Fig. 1 can be written in literal form, as follows:

$$P(\text{Top Event})\ \text{AND}\ P(\text{Event } a)\ P(\text{Gate} - 1 - 74) \quad (2)$$

$$P(\text{Gate} - 1 - 74)\ \text{OR}\ P(\text{Event } b)\ P(\text{Event } c)\ P(\text{Event } d) \quad (3)$$

The literal form can be adapted through use of Boolean equations, which allow the optimization of a fault tree and facilitation of the minimal cut sets characterization (Mohammad et al., 1999). Through use Boolean equations the Eqs. (2) and (3) can be rewritten as:

$$P(\text{Top Event}) = [P(\text{Event}A) \cap P(\text{Gate} - 1 - 74)] \quad (4)$$

$$P(\text{Gate} - 1 - 74) = [P(\text{Event } b) \cup P(\text{Event } c) \cup P(\text{Event } d)] \quad (5)$$

2.1.3. Minimal cut sets

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 (Jhon and Norman, 2011) it is possible to combine the Eqs. (4) and (5) in order to obtain a single expression. Through utilization of sets laws is possible to obtain an expression that can be processed more efficiently. For example, assuming that in Fig. 1 the events c and d are equal to the event a, we have:

$$P(\text{Top Event}) = P(\text{Event } a) \cap [P(\text{Event } b) \cup P(\text{Event } c) \cup P(\text{Event } d)] \quad (6)$$

$$P(\text{Top Event}) = P(\text{Event } a) \cap [P(\text{Event } b) \cup P(\text{Event } a) \cup P(\text{Event } a)] \quad (7)$$

$$P(\text{Top Event}) = P(\text{Event } a) \cap [P(\text{Event } b) \cup P(\text{Event } a)] \quad (8)$$

$$P(\text{Top Event}) = [P(\text{Event } a) \cap P(\text{Event } b)] \cup P(\text{Event } a) \quad (9)$$

$$P(\text{Top Event}) = P(\text{Event } a) \quad (10)$$

For extended Fault Trees the utilization of sets laws lead to a significant calculations reduction, without these simplifications whichever method used to calculate top event becomes extremely laborious and time-consuming, according to system complexity addressed.

2.2. 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 (Rudolph, 2009). 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.

2.2.1. Monte Carlo method

This technique allows the use of random numbers and mathematical statistical models to simulate real systems (Rudolph, 2009), 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 ($\beta = 1$) to the exponential distribution) (Rudolph, 2009), to be relatively simple and describe the weakest links of the failure mechanisms (Joe, 2012). 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 Eq. (11).

$$t = \mu . \ln[(1/(1 - F(t))]^{1/\beta}. \quad (11)$$

Such that:

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

Random numbers between 0 and 1 are used in the Monte Carlo simulation to find the cumulative distribution function of Weibull $F(t)$.

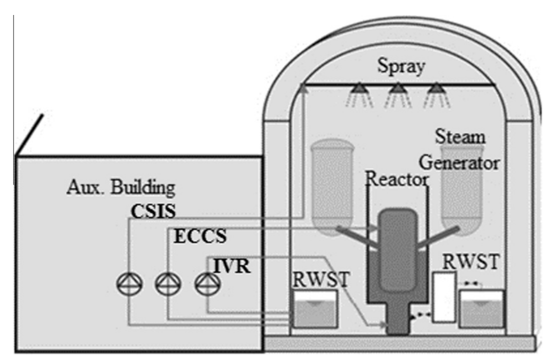

Fig. 2. PWR cooling systems (Kihwan et al., 2013).

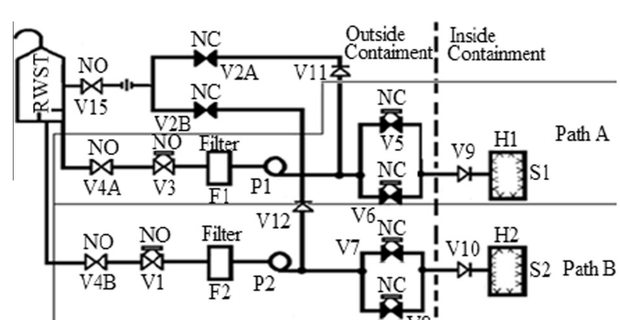

Fig. 3. Containment Spray Injection System Simplified Flow Diagram (Enrico, 2007).

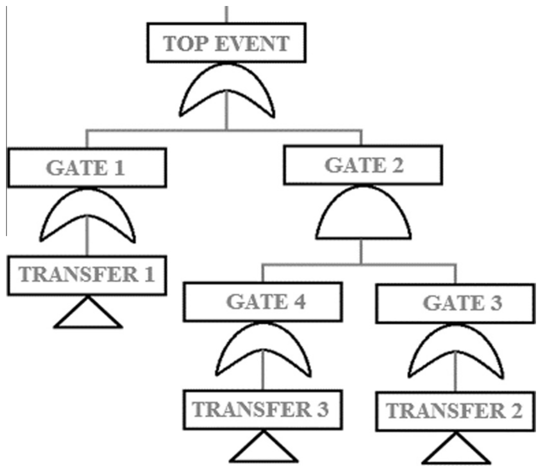

Fig. 4. Main Fault Tree of CSIS (U.S. Nuclear Regulatory Commission, 1975).

2.2.2. 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:

$$I_{FV} = \frac{P(TE|E_i = \langle E_i \rangle) - P(TE|E_i = 0)}{P(TE|E_i = \langle E_i \rangle)} \quad (12)$$

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.

3. Methodology application

The methodology application is directed to the Containment Spray Injection System of a PWR Nuclear Power Plant (see WASH

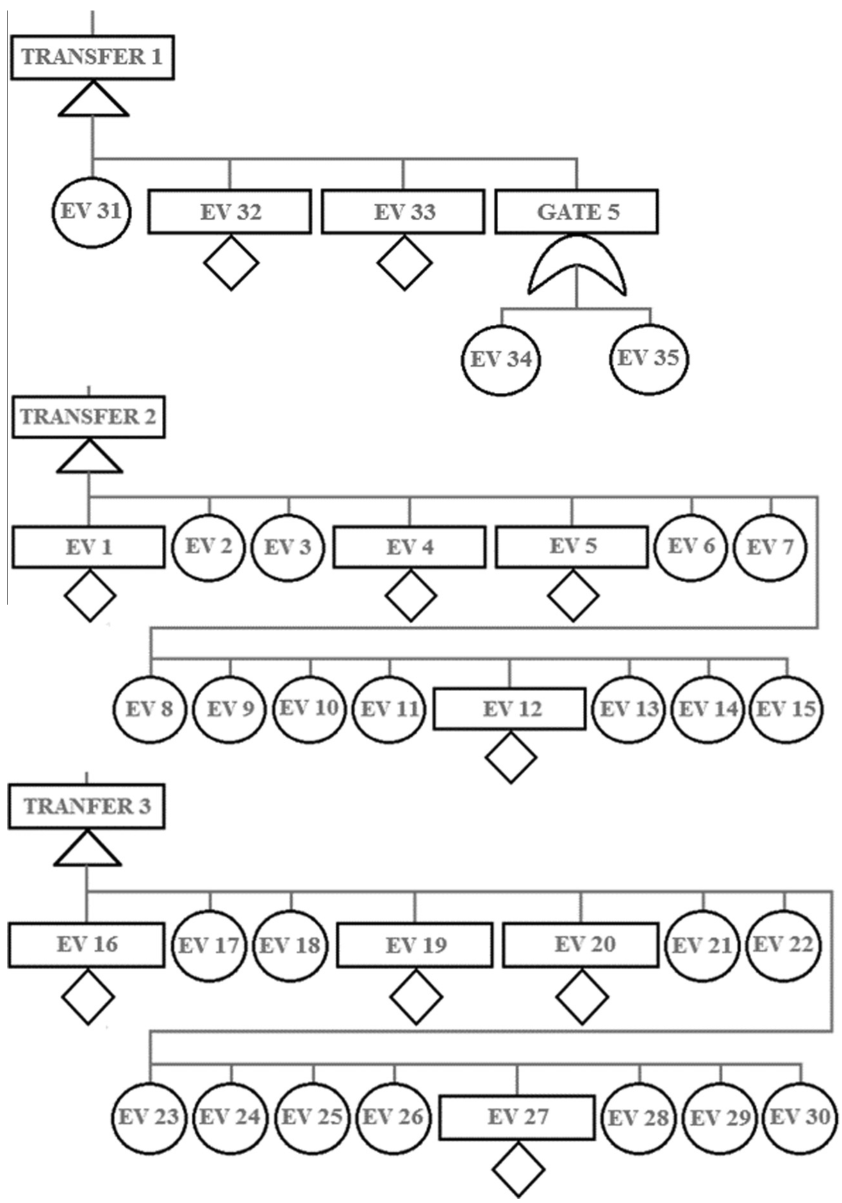

Fig. 5. Sub Trees of CSIS (U.S. Nuclear Regulatory Commission, 1975).

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

– 1400 U.S. Nuclear Regulatory Commission, 1975), 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) Hoagan, 2011. 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 3200 GPM from RWST to atmosphere through spray heads arranged in 360°. 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 (U.S. Nuclear Regulatory Commission, 1975). 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 Fig. 2 it is possible to see a simple scheme that shows the components distribution of the CSIS.

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 (U.S. Nuclear Regulatory Commission, 1975).

Table 2
Cut sets participation in system unavailability.

No. cut set	Frequency	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

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.

The failure probabilities and description of basic events from Fig. 5 are given in Table 1 (see WASH – 1400 U.S. Nuclear Regulatory Commission, 1975). The flagged events (Ev^*) indicate that are results from sub trees.

Table 3
System unavailability.

Basic event	Failure rate	System unavailability multiplicative factor			
		1× (Monte Carlo)	2×	5×	10×
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
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

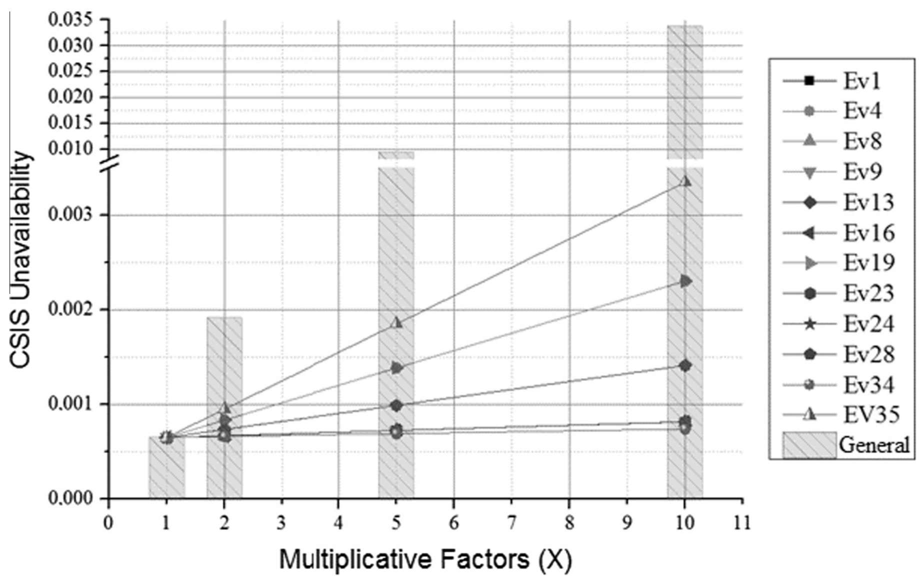

Fig. 6. Behavior of CSIS unavailability. (To better visualize the curves behavior, it was inserted stopping point on the vertical axis between the points 0.0035 and 0.008.).

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 Fig. 4 generates a hundred and seventy-two cut sets. (Table 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.

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

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

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

Through non-linear regression methods is possible to construct functions that govern the system behavior due to variation percentage of the most sensitive components to the aging. The functions obtained are of polynomial type of degree nine and are valid for the range of system failure, determined by multiplicative factors, for each of these basic event, such that:

$$F(x) = a_0 + a_1x + a_2x^2 + \ldots + a_9x^9 \quad (13)$$

where (x) represents the probability of component failure by operation time. The calculated values for the constant (CT) are shown in Table 4. As some events have a similar behavior, they can be presented as groups. The Group A comprises the events EV1, Ev8, EV9, EV16, EV23 and Ev24, the Group B the events Ev4 and EV19, the Group C the events Ev13 and Ev28, the Group D the event Ev34 and the Group E the event Ev35.

The values found for the unavailability of the system were as expected. Due to the positioning of basic events and values of

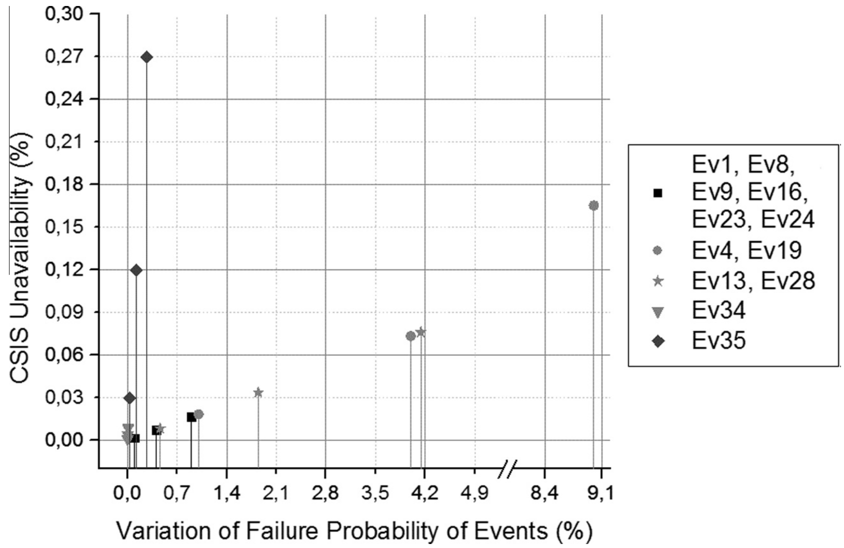

Fig. 7. Progression percentage of unavailability of the CSIS. (To better visualize the curves behavior, it was inserted stopping point on the horizontal axis between the points 5.3 and 8.0.).

Table 4
Constant functions.

CT	Groups of basic events				
	Group A	Group B	Group C	Group D	Group E
a_0	−3.47716E−05	1.00000E−05	−3.10635E−06	−1.69965E−17	−2.86409E−05
a_1	0.01979E+00	0.01838E+00	0.01844E+00	1.00000E+00	1.00450E+00
a_2	−0.02185E+00	−4.87320E−16	−1.91252E−04	−5.61694E−11	−0.26980E+00
a_3	0.17686E+00	3.00973E−16	3.36494E−04	3.55331E−08	7.27865E+00
a_4	−0.82092E+00	−1.14584E−16	−3.39536E−04	−1.35843E−05	−1.12613E+02
a_5	2.28691E+00	2.79707E−17	2.05625E−04	0.00328E+00	1.04578E+03
a_6	−3.86952E+00	−4.37464E−18	−7.56353E−05	−0.50279E+00	−5.89798E+03
a_7	3.89110E+00	4.22362E−19	1.65341E−05	4.73606E+01	1.97696E+04
a_8	−2.13811E+00	−2.28483E−20	−1.97507E−06	−2.49660E+03	−3.62104E+04
a_9	0.49444E+00	5.28409E−22	9.92893E−08	5.63118E+04	2.79120E+04

failure probability, the system may be affected in the same way for different events. The determination of the most sensitive components are obtained through the measure of Importance Fussell–Vesely, however, they can also be identified by the impact they have in the system.

The multipliers are presented as a useful tool when it ignores the time the events occurred. Its use allows to consider the time as an implicit factor of the system to be analyzed. The values of multiplicative factors were chosen arbitrarily, such that its use allowed to verify the system behavior in a period of operation not determined.

The application of the theoretical definition of the aging process was done using the multiplicative factors. Without the application of these factors would not be possible to see how the components of the CSIS would impact it, for example, analyzing the system unavailability (Table 3) due to the influence of its components without changes in their probabilities of failure (1× (Monte Carlo)) is not possible to estimate how the aging could sensitize it. With the use of the factors it is possible to see the impact that each component cause in the system (Table 5), in other words, it becomes possible to estimate how the unavailability of the system varies when the probability of failure of its components suffer an increase.

Through the analysis of the aging process, considering the contribution of the events for the variation percentage of the system unavailability, it becomes possible to analyze how the system will be behave considering the time operating of its component, so that time can be estimated by means that differ from the need to record the behavior of the components throughout its operating time.

Table 5
Percentage change of the system unavailability.

Event	Failure rate of components (%)			System unavailability (%)		
	2×	5×	10×	2×	5×	10×
Group A	0.100	0.400	0.900	0.00184	0.00736	0.01655
Group B	1.000	4.000	9.000	0.01839	0.07353	0.16543
Group C	0.460	1.840	4.140	0.00846	0.03383	0.07613
Group D	0.001	0.004	0.009	0.00100	0.00400	0.00900
Group E	0.030	0.120	0.270	0.02999	0.11993	0.26993

The determination of the functions that govern the system behavior is a good alternative to estimate quickly how a component can impact it. This paper made an estimate for a small range of system operation, however, it is possible to determine functions that show how the components influence a system from its creation to its full stop by failure. This determination can estimate relevant data for the analysis of system failure at any period of its operation time.

4. Conclusions

The term aging for nuclear power 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. 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 (Figs. 6 and 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 methodology, the estimate of system failure would depend on a detailed historical of the behavior of the components of the system for a considerable period of time.

For the future, it is strongly recommend a study that addresses how the human reliability and the advanced state of corrosion impact in the failure probability of a system in a nuclear power plant.

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References

David, M., 2004. Practical Hazops, Trips and Alarms. Elsevier, Oxford, United Kingdom.
Enrico, Z., 2007. An Introduction to the Basics of Reliability and Risky Analysis. World Scientific Publishing, New Jersey, United States of America.
Hoagan, P., 2011. Safety and Risk Modeling and Its Applications. Springer, London, United Kingdom.
Jhon, C.L., Norman, J.M., 2011. Risk and Safety Analysis of Nuclear Systems. John Wiley & Sons, New Jersey, United States of America.
Joe, W.M., 2012. Reliability Physics and Engineering. Springer, New York, United States of America.
Kihwan, L., Kang-Heon, L., Jeong, L.L., Young, H.J., Phill-Seung, L., 2013. A new design concept for offshore nuclear power plants with enhanced safety features. Nucl. Eng. Des. 254, 10–11.
Mohammad, M., Mark, K., Vasiliy, K., 1999. Reliability Engineering and Risk Analysis. Marcel Dekker, New York, United States of America.
Rudolph, F.S., 2009. Handbook of Reliability, Availability, Maintainability and Safety in Engineering Design. Springer, London, United Kingdom.
U.S. Congress, Office of Technology and Assessment, 1993. OTA-E-575: Aging Nuclear Power Plants: Managing Plant Life and Decommissioning, Washington DC, pp. 1–187.
U.S. Nuclear Regulatory Commission, 1975. WASH-1400: Reactor Safety Study, Washington DC, pp. II-111–II-115.
U.S. Nuclear Regulatory Commission, 1981. NUREG-0492: Fault Tree Handbook, Washington DC, pp.IV-1–IV-2.
U.S. Nuclear Regulatory Commission, 1987. NUREG-1144 Rev. 1: Nuclear Plant Aging Research (NPAR) Program Plan, Washington DC, pp. 1–135.
U.S. Nuclear Regulatory Commission, 1990. NUREG-5507: Results from the Nuclear Plant Aging Research Program: Their Use in Inspection Activities, Washington DC, pp. 1–81.
U.S. Nuclear Regulatory Commission, 1991. NUREG-1144 Rev. 2: Nuclear Plant Aging Research (NPAR) Program Plan, Washington DC, pp. 1–175.
U.S. Nuclear Regulatory Commission, 2000. NUREG-1723: Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, Washington DC, pp. 1–551.
U.S. Nuclear Regulatory Commission, 2003. NUREG-1782: Safety Evaluation Report Related to the License Renewal of Unit 1, Washington DC, pp. 1–502.