INSTITUTO DE ENGENHARIA NUCLEAR

RT-IEN-15/2002

The Use of Fuzzy Logic Control Through Risk-Informed Methods to Establish Inservice Inspection Priorities For Nuclear Components

por

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Agosto/2002

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Identificação: RT-IEN-15-2002	Nº de páginas: 22	Tipo de Divulgação: Irrestrita (x) Restrita ( )	Divulgar para: IEN	Localização: DIRE/SETR
Publicação externa associada (congresso/periódico): Expert Systems with Applications, Elsevier.				
Palavras chave: *Fuzzy logic, ANFIS, probabilities, Core Damage Frequency, Risk.*				

Resumo:

Este trabalho descreve o uso do controle lógico nebuloso através do método informado no risco para estabelecer prioridades de inspeção em serviço para componentes nucleares com *Adaptive Neural Fuzzy Inference system* (ANFIS). Para efeito de demonstração da metodologia proposta aqui foram utilizadas informações de especialistas e os resultados de *um plant-specific probabilistic risk assessment* (PRA) para identificar prioridades de inspeção para os componentes. Os sistemas de grande porte utilizados foram o *auxiliary feedwater*, *low-pressure injection*, e o *reactor coolant systems* da instalação nuclear americana *Surry Nuclear Power Station Unit 1* (Surry-1). Três exemplos foram desenvolvidos e os resultados estimados quando comparados aos resultados do trabalho original apresentaram importantes conclusões.

Abstract:

This work describes the use of fuzzy logic control through risk-informed method to establish inservice inspection priorities for nuclear components with Adaptive Neural Fuzzy Inference system (ANFIS). For demonstration of the methodology proposed here, the risk importance of components, the probabilities of components failures, estimated by using an expert judgment elicitation process and the core damage frequency of each component failure obtained from plant-specific probabilistic risk assessment (PRA) results were used to identify in-service inspection (ISI) priorities for components. The specific systems addressed in this work are the auxiliary feedwater, low-pressure injection, and reactor coolant systems selected from the Surry Nuclear Power Station Unit 1 (Surry-1). Three examples were developed to illustrate the methodology proposed here and the results obtained when compared with the reference work lead to important conclusions.

Emissão		Nome	Rubrica	Data
Data: 20/08/2002	Elaboração:	Antonio Cesar Ferreira Guimarães		20/08/2002
Divisão:DIRE	Revisão:	Orlando J. A Gonçalves Filho		20/08/2002
Serviço:SETER	Aprovação :	Orlando J. A. Gonçalves Filho		20/08/2002

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1. INTRODUCTION

As part of the Evaluation and Improvement of Nondestructive Evaluation (NDE) reliability for Inservice Inspection of Light Water Reactors Program sponsored by U.S. Nuclear Regulatory Commission (NRC), the Pacific Northwest National Laboratory (PNNL) has developed and applied a method using risk-informed techniques for inservice inspection (ISI) plans of nuclear power plants. The method uses probabilities of components failures (estimated by using an expert judgment elicitation process) and plant-specific probabilistic risk assessment (PRA) results to identify ISI priorities for components. Worksheets to guide the analysis were initially formulated using plant system drawings and other plant-specific information. The Standard Review Plan information developed by NRC was used in determining the effects of system and component failures. To ensure that the plant models were as realistic as possible, visits at the plant were conducted for plant systems walkdowns and discussions were held with plant operational and technical staff. Participation of Virginia Electric Power Company staff was an essential part of the pilot study (Vo, et al., 1997). The results of risk-informed component prioritization included the estimated rupture probabilities for the components of the systems analyzed. On the basis of core damage frequency, the calculated contributions of component failures to core damage frequency range from about 1.0E-12 to 6.0E-06 per plant year. The cumulative risk contribution for all of the components considered was estimated to be about 1.8E-05 per plant year. The total estimated risk is dominated by failures of the auxiliary feedwater system components within the system components (60% of the total estimated risk). The low-pressure injection system components (39%), and then other various components within the reactor coolant system (1%) follow this risk.

Also, sensitivity analysis was performed to address the changes in component rankings using the upper/lower estimates of component rupture probabilities. The results indicated no significant changes in component risk contribution rankings. Sensitivity analyses were also performed to determine the core damage frequency contribution due to component failures by indirect effects (pipe whip, jet impingement effects, etc.) The results indicate that contributions from the indirect effects were negligible.

This context is part of report that is a revision of the earlier report (NUREG/CR-6181), which incorporates recent plant-specific information and improved risk-informed calculations tools. Since that, the NUREG/CR-6181 Rev. 1, provides a preferred methodology, it supersedes the earlier NUREG/CR-6181 report published in August 1994.

Inside this paper a new methodology for this theme was proposed and used for estimate the core damage frequency starting from the probabilities of failures estimated by experts, and then the contribution for the fuzzy risk-informed component prioritization. In (Guimarães, 2002) and (Uhrig et al., 1997), this approach has applied so well.

In the next item, the ANFIS approach applied in this paper will be described and the item 3 will be presented an application with this methodology with theme. The item 4, the results obtained with this methodology proposed here will be presented. And, item 5, the conclusions will be described.

2. METHODOLOGY ANFIS

This section describes the methodology, which was used to perform the estimated core damage frequency and the risk-ranking process. This is a new methodology in this type of theme.

An ANFIS is an FIS (Fuzzy Inference System) that can be trained with a backpropagation algorithm to model some collection of input/output data. Allowing the system to adapt provides the fuzzy system with the ability to learn the input/output relationships embedded in the collected data. The ANFIS network structure facilitates the computation of a gradient vector that relates the reduction of an error function to a change in the parameters of the FIS. Once this gradient vector is obtained, a number of optimization routines can be applied to reduce the error between the actual and desired outputs. In the neural network literature, this process is called learning by example (Uhrig et al., 1997). The ANFIS described here uses the Sugeno-style fuzzy model (also known as the TSK fuzzy model) proposed by Tagaki and Sugeno (1985) and Sugeno and Kang (1988). In Takagi and Sugeno (1985) proposed to use the following fuzzy IF-THEN rules:

$$L^{(l)} : IF\ \ x_1\ \ is\ F_1^{\,l}\ \ and\ \ \ldots\ \ and\ \ x_n\ \ is\ F_n^{\,l},$$

$$THEN\ \ \ y^l = c_0^{\,l} + c_1^{\,l}\, x^1 + \ldots + c_n^{\,l}\, x_n \qquad (2.1)$$

Where F_i^l are fuzzy sets, c_i are real-valued parameters, y_l is the system output due to rule $L^{(l)}$, and $l = 1,2, \ldots, M$. That is, they considered rules whose IF part is fuzzy but whose THEN part is crisp – the output is a linear combination of input variables. For a real-valued input vector $\underline{x} = (x_1, \ldots, x_n)^T$, the output $y(\underline{x})$ of Takagi and Sugeno's fuzzy system is a weighted average of the y^l's:

$$Y\,(\underline{x}) = (\sum{}^{M}_{l=1}\ w^l y^l\,) \,/\, (\sum{}^{M}_{l=1}\, w^l\,) \qquad (2.2)$$

Where the weight w^l implies the overall truth value of the premise of rule $L^{(l)}$ for the input and is calculated as:

$$W^l = \prod{}^{n}_{i=1} \mu_{F\,l\,i}\,(x_i) \qquad (2.3)$$

The configuration of Takagi and Sugeno's fuzzy system is shown in Figure 1.

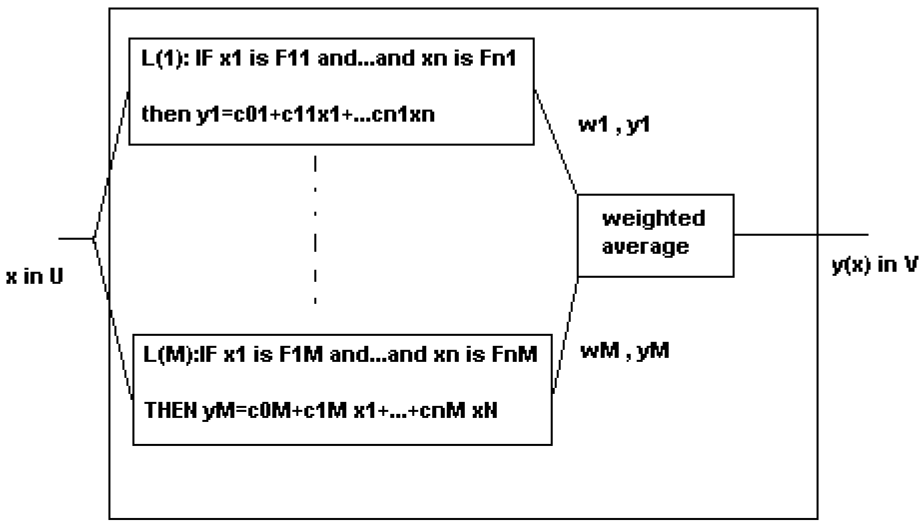

Figure 1 – Basic configuration of Takagi and Sugeno's fuzzy system.

MATLAB6's software package and its associated fuzzy logic toolbox (2000) were used to create the ANFIS – based risk importance components. MATLAB6's ANFIS support first-order Sugeno systems that have a single output and unity weights for each rule.

3. APPLICATION USING ANFIS

The Surry-1 systems selected for this re-analysis study to evaluate the new methodology were the primary pressure boundary system, the front-line safety systems, and certain important support systems. These were the auxiliary feed-water (AFW), low-pressure injection (LPI), and the reactor coolant (RCS) systems. Detailed descriptions can be found in the Surry-1 FSAR (Final Safety Analysis Report). General assumptions, total of 10, were used for the analysis (Vo et al., 1997), but only one will be presented here because is the most important of ten for this work:

- *Core damage frequency was used as the bottom-line risk measure to put priority in plant system components.*

The following sources of information were used to put priority in components for inspection: (i) the component failure probabilities estimated from expert judgment elicitation (Vo et al. 1990, 1991, and 1993), and (ii) Surry-1 PRA (NUREG/CR-4550 Bertucio and Julius 1990).

Within the three systems analyzed, there are approximately 200 individual pipe segments. By assuming that identical components in identical trains within the same system have the same failure probabilities and consequences, this total is reduced to approximately 100. For ranking purpose, components within the same train can be further grouped, based on major discontinuities (e.g., between pumps and major valves). This resulted in 24 major groups within the systems analyzed. Figure 2 show the results of the risk-informed ranking of major components within three selected systems at Surry-1, based on the contributions of component failures to core damage frequency, where CDF is the Core Damage Frequency, RupFreq is the Rupture Frequency:

System component	Rank	RupFreq	CDF
LP1 – pipe segment between containment isolation valve (inside) and cold leg injection	C1	2.65E-05	5.96E-06
AFW - AFW isolation valve to SG	C2	2.63E-04	3.31E-06
AFW - pipe segment between containment isolation and SG isolation valves	C3	5.27E-05	2.91E-06
AFW - AFW MDP Discharge line	C4	4.33E-05	2.39E-06
AFW - CST supply line	C5	1.84E-05	1.01E-06
LPI - LPI sources (RWST, Sump), Supply, Line.	C6	2.30E-05	6.85E-07
AFW - FW MDP Suction Line	C7	1.00E-05	5.60E-07
AFW - pipe segment from Unit 2 AFW pumps	C8	3.00E-06	3.50E-07
LPI - pipe segment between containmnet isolation valve (inside) and hot leg	C9	1.33E-05	2.84E-07
AFW - AFW TDP Suction line	C10	5.00E-06	2.78E-07
AFW - AFW TDP Discharge line	C11	5.00E-06	2.76E-06
LPI - LPI pump suction line	C12	7.60E-06	2.02E-07
RCS - Pressurizer Spray Line	C13	2.70E-05	5.15E-08
LPI - Pipe segment between pump discharge and containmnet isolation valves	C14	1.29E-05	4.51E-08
RCS - Pressurizer Relief/Safety line	C15	8.40E-06	2.86E-08
AFW - main steam to AFW pump turbine drive	C16	1.51E-05	1.48E-08
RCS - pipe segment between RPV and loop step valve (hot leg)	C17	3.00E-06	1.19E-08
RCS - pressurizer surge line	C18	1.60E-06	6.38E-09
RCS - pipe segment between SG and RCP	C19	9.00E-07	2.84E-09
RCS - pipe segment between loop stop valve and SG (hot leg)	C20	4.00E-07	1.89E-09
RCS - pipe segment between loop stop valve and RPV (cold leg)	C21	3.00E-07	1.20E-09
RCS - pipe segment between RCP and loop stop valve (cold leg)	C22	2.00E-07	9.31E-10
LPI - Pipe segment between containment isolation valves	C23	1.80E-06	9.00E-10
AFW - pipe segment from emergency makeup system and from fire main to AFW.	C24	8.06E-06	<1.0E-12

Figure 2 – Risk importance for components of selected systems at Surry-1

3.1 – Variables of Fuzzy Inference Systems

Using the rupture frequency values showed in Figure 2, we could display them for easy interpretation of input pattern. Using ANFIS to generate the fuzzy system FIS, was defined some fuzzy parameters:

Name: 'Risk'
    type: 'sugeno'
    andMethod: 'prod'
    orMethod: 'probor'
    defuzzMethod: 'wtaver'
    impMethod: 'min'
    aggMethod: 'max'
    input: [1x1 struct]
    utput: [1x1 struct]
    rule: [1x100 struct]

All this parameters can be change of value using the ANFISeditor of MATLAB6. It was used a PC Intel Pentium II 500MHz, 128 Mbytes, to process the network ANFIS. The results come so fast in this configuration.

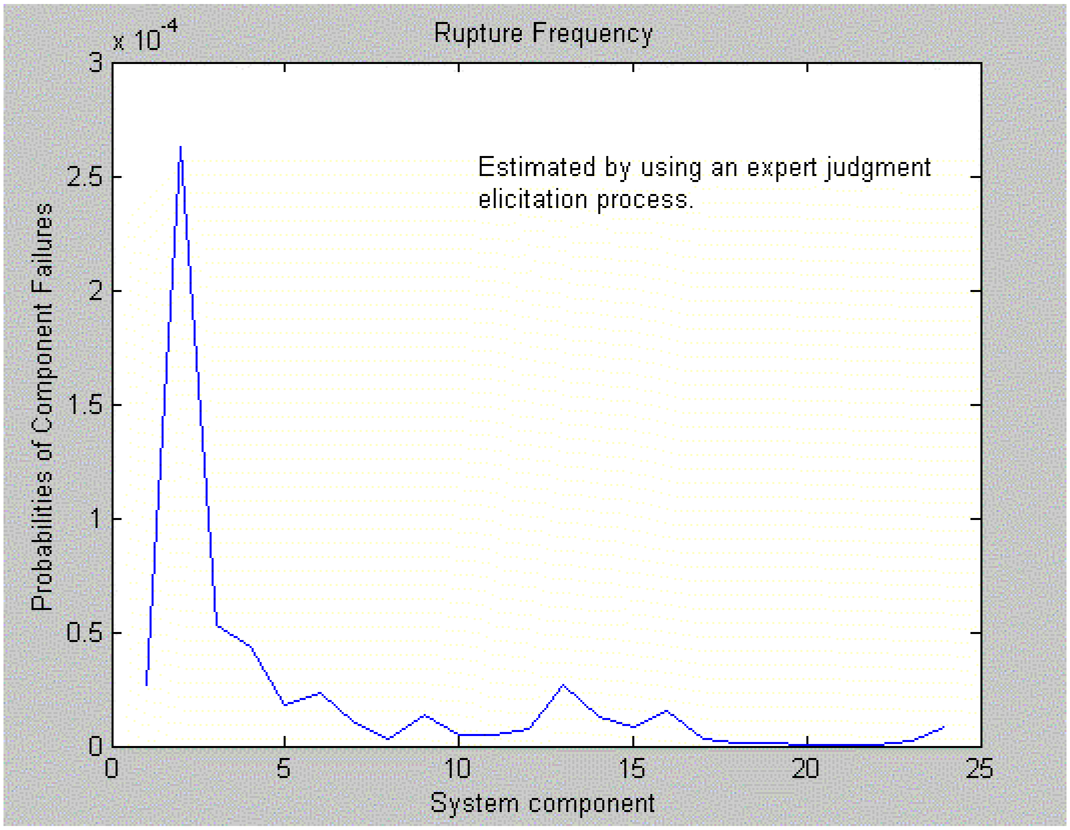

Figure 3 – Rupture Frequency (input variable)

3.2 – Risk Importance for Components

Using the core damage frequency values, showed in Figure 2, we can display them for easy interpretation of output pattern.

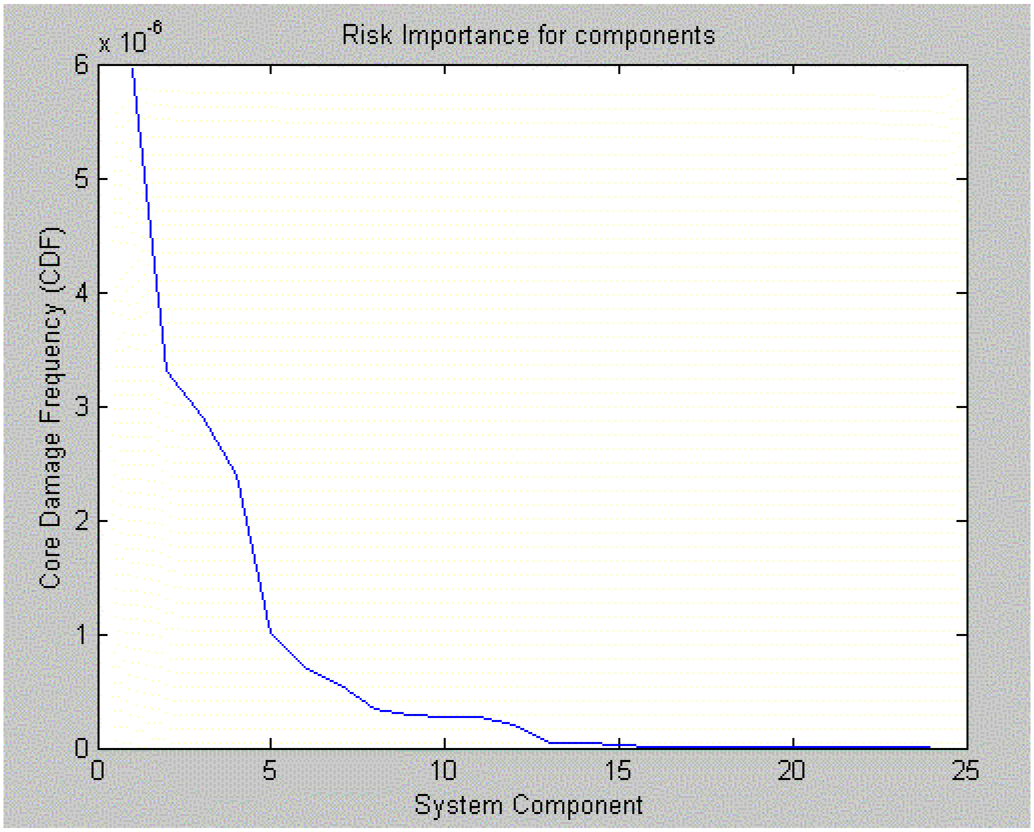

Figure 4 - Risk Importance for Components (output variable)

Using the FIS generated, three sample case were illustrated for CDF estimated: (i) comparative results between NUREG/CR-6181 Ver.1 calculations and ANFIS methodology estimated (ii) “reviewed” situation where need modifications in values of rupture frequency probabilities, and (iii) “reduced” situation in rupture frequency probabilities due to “Inspection Program Development” implemented.

4. RESULTS

In the Figure 5, the comparative results are showed between NUREG/CR - 6181 calculation and the estimated by ANFIS methodology.

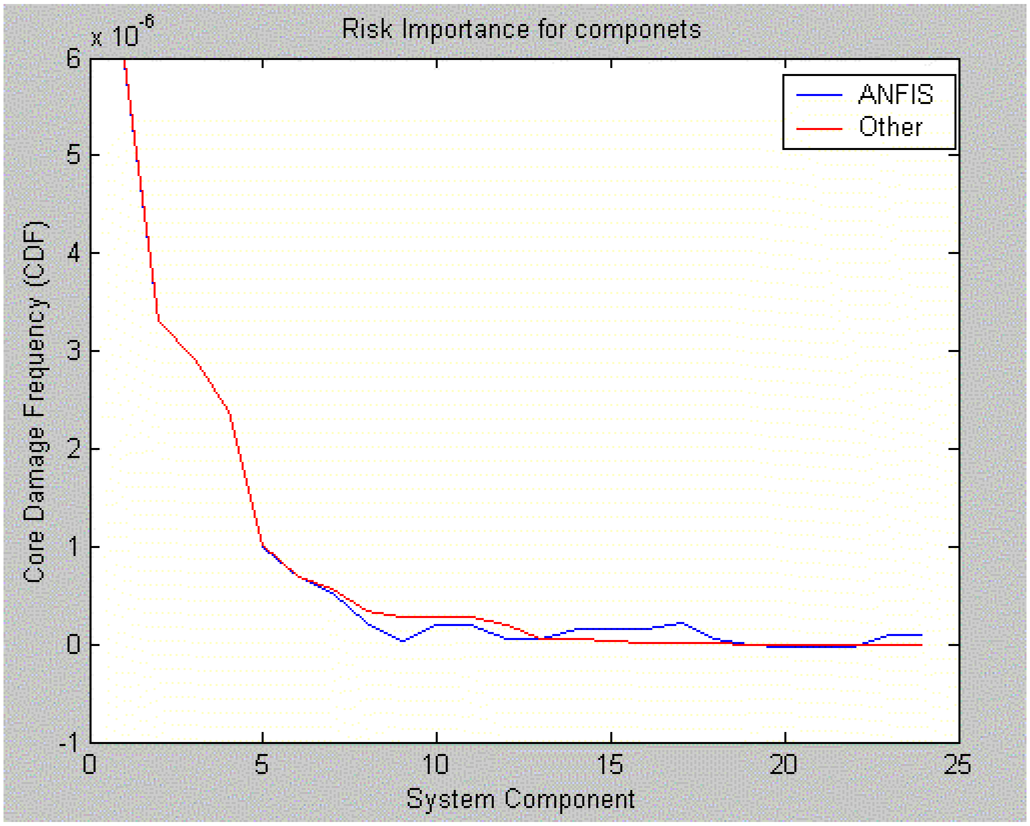

Figure 5 – Comparative results between observed and estimated with ANFIS.

For illustration effect, in the Figure 6, only in the first example, comparative results for “cumulative risk contribution” between calculated in NUREG/CR - 6181 and the estimated with ANFIS methodology was presented.

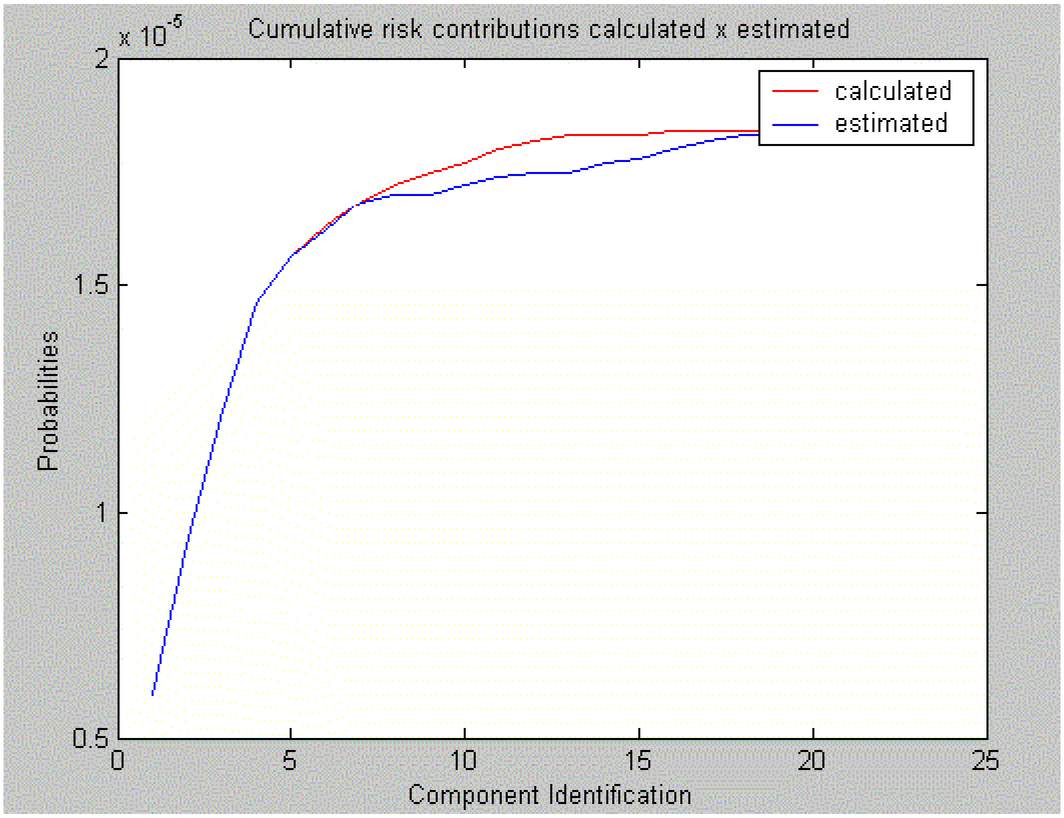

Figure 6 – Cumulative risk contribution calculated x estimated.

In the Figure 7, a new elicitation process was defined, sample case two, for ANFIS estimate the "Core Damage Frequency" contribution of each component for "Risk Importance for Components".

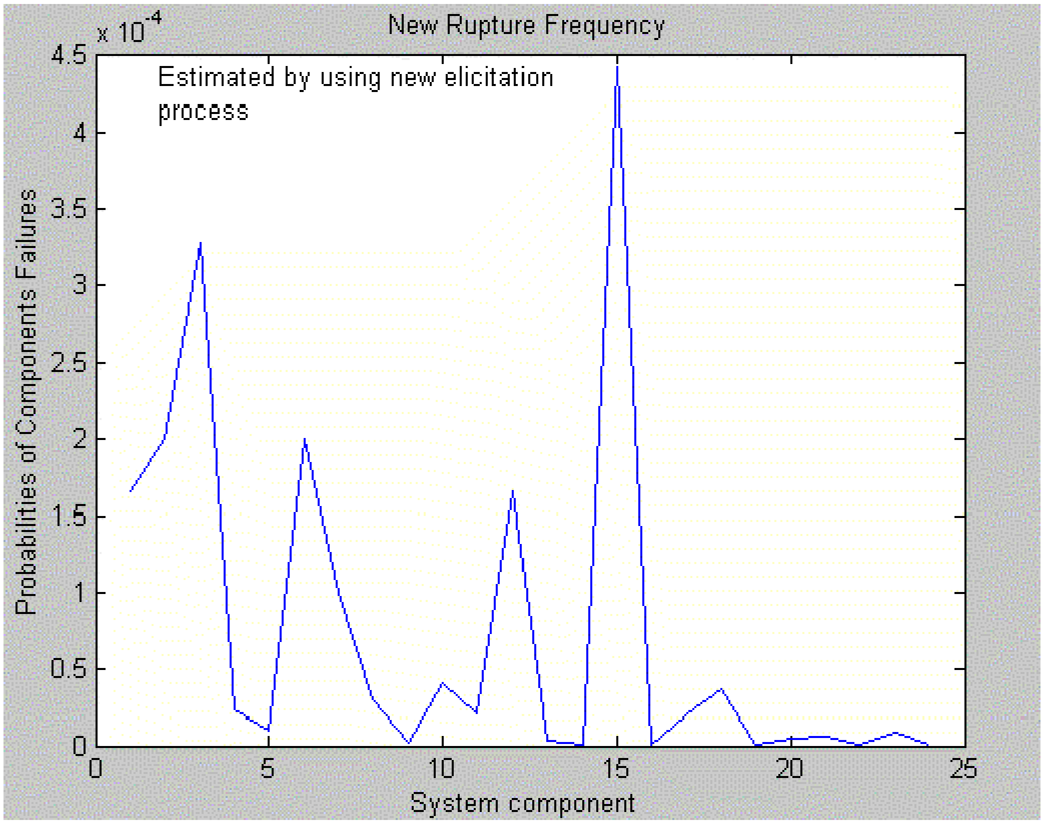

Figure 7 - New values for probability rupture.

In Figure 8, the “risk importance for components” was estimated using a sample case two, “reviewed”, with new values of probabilities for rupture frequency. Repair that in this Figure 8 the components are not ranked yet.

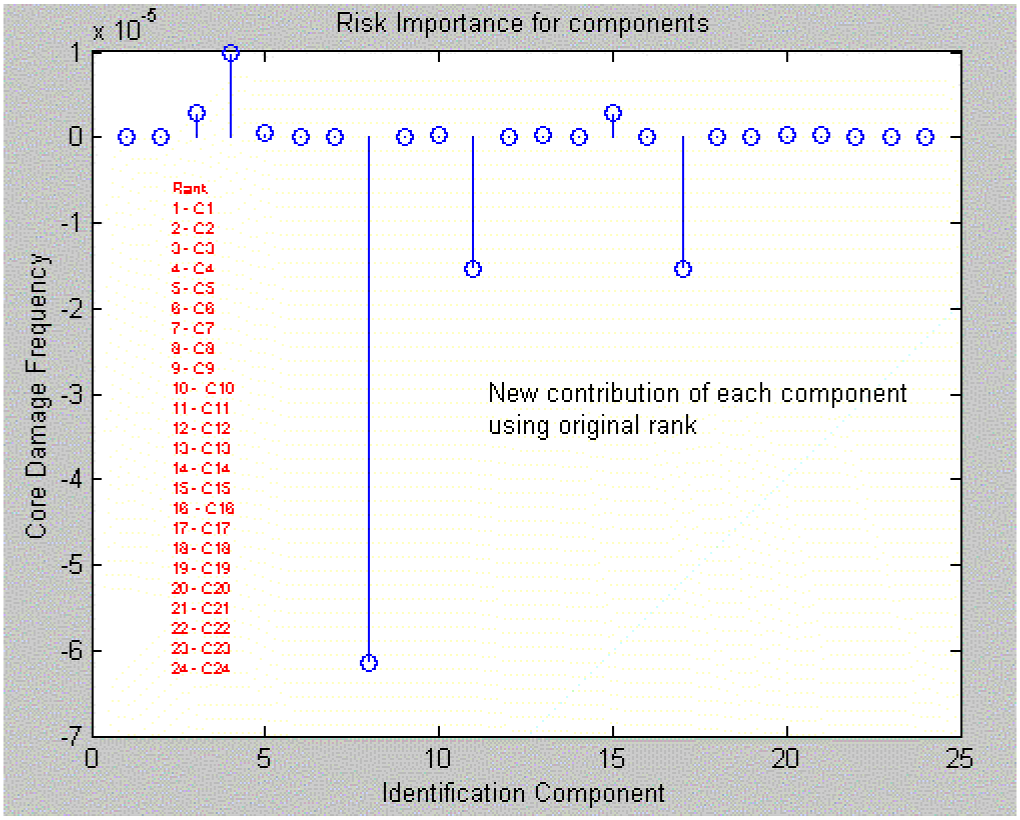

Figure 8 – Results for Risk Importance for components.

In Figure 9, a new rank is defined for components combined with the reduced probability values.

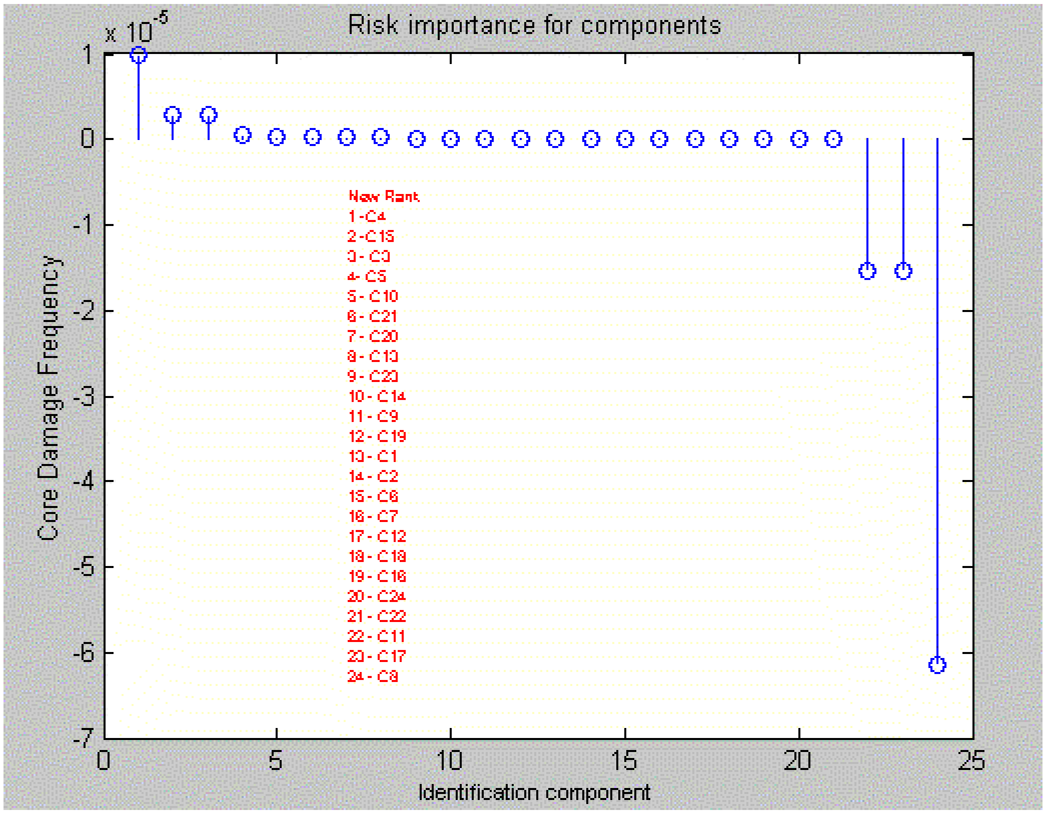

Figure 9 – Risk Importance for components Ranked.

With new results for CDFs a new rank for component was realized.

Rank	RupFreq	CDF	CDFest	NRupFre	NCDFestOrd	NRank
C1	2.65E-05	5.96E-06	5.96E-06	1.65E-04	9.87E-06	C4
C2	2.63E-04	3.31E-06	3.31E-06	2.00E-04	2.98E-06	C15
C3	5.27E-05	2.91E-06	2.91E-06	3.27E-04	2.98E-06	C3
C4	4.33E-05	2.39E-06	2.39E-06	2.33E-05	5.10E-07	C5
C5	1.84E-05	1.01E-06	1.01E-06	1.00E-05	3.00E-07	C10
C6	2.30E-05	6.85E-07	6.85E-07	2.00E-04	2.70E-07	C21
C7	1.00E-05	5.60E-07	5.06E-07	1.00E-04	2.40E-07	C20
C8	3.00E-06	3.50E-07	1.4E-07	3.00E-05	1.80E-07	C13
C9	1.33E-05	2.84E-07	1.4E-07	1.33E-06	1.50E-07	C23
C10	5.00E-06	2.78E-07	2.89E-07	4.03E-05	3.00E-08	C14
C11	5.00E-06	2.76E-06	2.88E-07	2.00E-05	3.00E-08	C9
C12	7.60E-06	2.02E-07	7.8E-08	1.65E-04	1.00E-08	C19
C13	2.70E-05	5.15E-08	5.3E-08	3.70E-06	0.00E+00	C1
C14	1.29E-05	4.51E-08	2.01E-07	1.29E-06	0.00E+00	C2
C15	8.40E-06	2.86E-08	8.4E-08	4.41E-04	0.00E+00	C6
C16	1.51E-05	1.48E-08	1.9E-08	5.10E-07	0.00E+00	C7
C17	3.00E-06	1.19E-08	1.4E-07	2.00E-05	0.00E+00	C12
C18	1.60E-06	6.38E-09	5.6E-08	3.60E-05	0.00E+00	C18
C19	9.00E-07	2.84E-09	5E-09	9.24E-07	-1.00E-08	C16
C20	4.00E-07	1.89E-09	-1.4E-08	4.35E-06	-2.00E-08	C24
C21	3.00E-07	1.20E-09	-1.7E-08	6.06E-06	-2.00E-08	C22
C22	2.00E-07	9.31E-10	-1.8E-08	3.45E-07	-1.54E-05	C11
C23	1.80E-06	9.00E-10	7.6E-08	8.83E-06	-1.54E-05	C17
C24	8.06E-06	<1.0E-10	6.7E-08	1.06E-07	-6.14E-05	C8

Figure 10 - Numerical values for estimated and new estimated for risk importance and rank. (sample case i and ii)

In Figure 10, all values for example (i) and example (ii) are presented. The **Rank**, **RupFreq**, and **CDF** are the values from NUREG/CR for ANFIS estimated analysis. The **NrupFre** values are the new rupture frequency values for new analysis with ANFIS, without PRA results. The **NCDFestOrd** values are the new values estimated with ANFIS methodology, and a new component rank, **Nrank**, are defined with new **NCDFestOrd** ranked values.

For illustration effect, a new calculation with ANFIS methodology, sample case three, was proposed using now, reduced probability of each component. This may be possible as recommendation of Inspection program development for plant analysis.

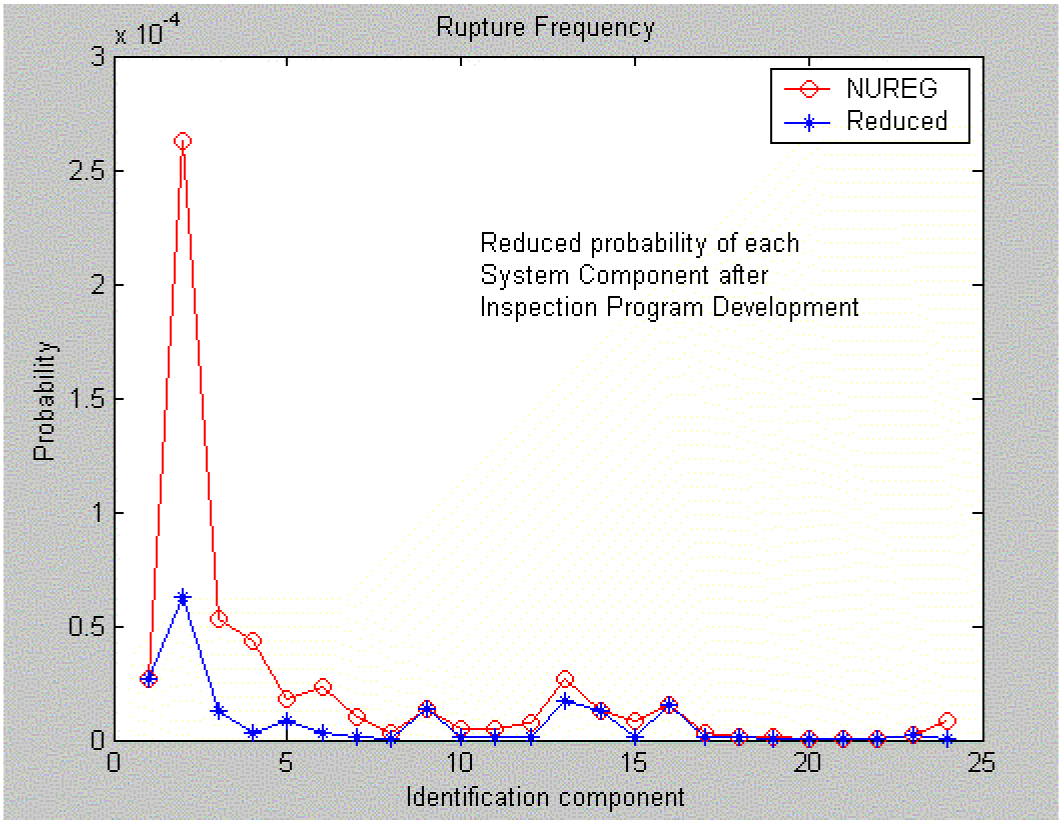

Figure 11 – New data for Rupture Frequency

In the Figure 11, a new set of data for probabilities after Inspection Program Development implemented on plant with reduced probability is waiting, and then the ANFIS system can obtain the contribution of each component for core damage frequency.

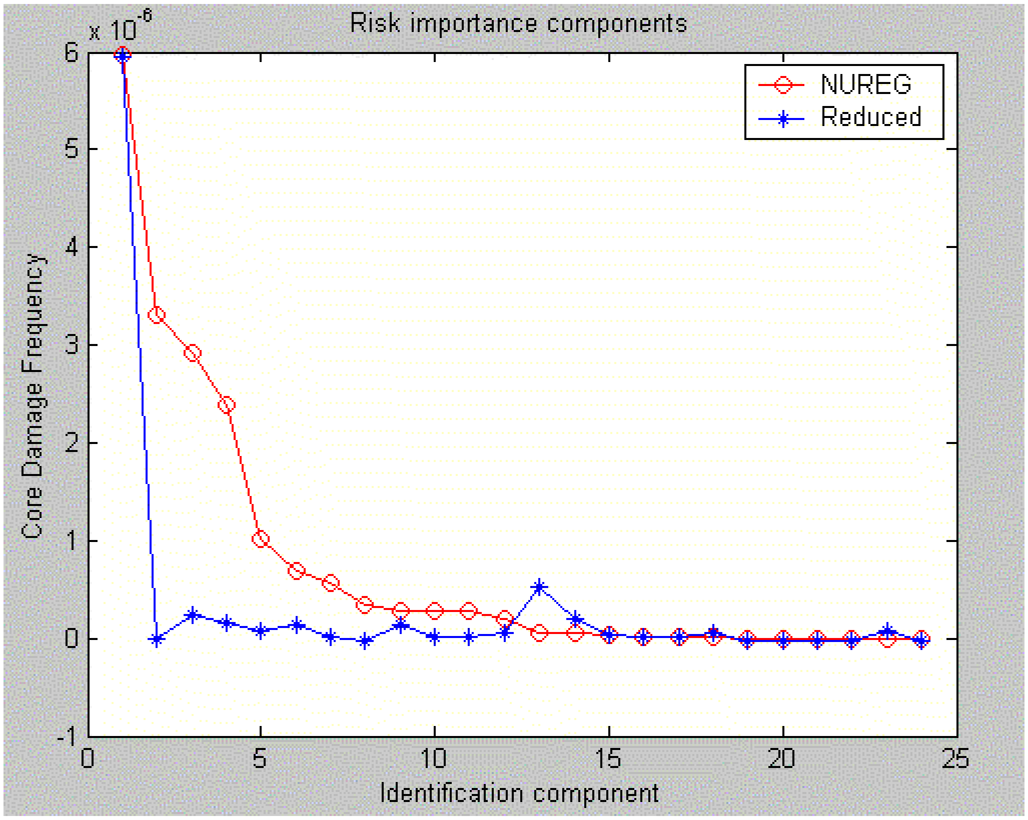

Figure 12 – Risk importance component for reduced rupture component probabilities

In Figure 12, comparative study showing the risk importance components in the case of NUREG/CR – 6181 and the risk estimated with ANFIS methodology, a sample case using reduced probabilities for rupture frequency.

5. CONCLUSION

In this paper a new methodology was proposed using the Fuzzy Logic Control applied to estimation of Core Damage Frequency from Rupture Frequency for Risk Importance for Components. For this, another reference important study (Vo, et al., 1997) was used and the data contained in this work used in this paper for news estimations.

For the systems selected in that study, a detailed component level prioritization was performed. Simplified Piping and Instrumentation Diagram (P&ID) drawings for each system of interest and worksheets were developed to support the analysis (see Vo et al., 1997). The P&ID drawings were to identify the pipe segment boundaries. The worksheets contained information specific to a component, such as the failure probability and consequences of component rupture. The consequence type is categorized as being a component failure that causes system degradation; an initiating event; or a combination of system degradation and an initiating event. To compute the contribution to core damage frequency of a component, the plant PRA was used to find a basic event whose failure would have the same effects as a component rupture. In that analysis (Vo et al., 1997), the risk increase for that basic event was used to measure the contribution to core damage frequency of a component rupture. The risk increases for the basic events were calculated by setting the failure probability for the events to one, and the computing the new core damage frequency. In a case of news examples, “revision” or “reduction” one, modifications in probabilities of rupture frequency occur. With ANFIS methodology, a plant PRA was not used and then all process to computing the new core damage frequency is not necessary.

As the PRA is not used as input to the core damage frequency (CDF) calculation, the postulated consequences of the failure are not identified. Then based on the not necessary identified consequences, the PRA model wasn't manipulated to obtain the required information. The consequence analysis considered from both direct effects and indirect effects are not realized. Because the consequences can vary and the correct PRA and failure probability information is necessary for the CDF calculation in Vo (1997), the process requires different manipulations for each type of consequence. Different equations were developed to ensure the proper calculation for each type of consequence in Vo (1997). The risk increase values are combined with the results of the component failure probability/rate to obtain core damage frequency for each component. Depending upon the type of consequence; one of the three equations (as shown in Vo, et al., 1997) was used to compute the component or pressure boundary core damage frequency. With ANFIS methodology all this process is not necessary.

The methodology and results of the present work represent a contribution of an approach to risk-informed inservice inspection and should not be used as a basis for actual changes to any plant inservice inspection plans. A detailed fracture mechanics calculations, the plant-specific evaluation, and up to date plant information must be developed as well a set of checking and testing data available to validate this study as described in the presented paper.

6. REFERENCES

Bertucio, R. C., & J. A. Julius. 1990. *Analysis of Core Damage Frequency: Surry, Unit 1 internal Events*. NUREG/CR-4550, Sandia National Laboratories, Albuquerque, New Mexico.

Guimarães, A. C. F. 2002, "A New Methodology for the Study of FAC Phenomenon based on a Fuzzy Rule System." ANE – Annals of Nuclear Energy (to be published).

Mamdani, E. H. 1974, "Applications of fuzzy algorithms for simple dynamic plant", Proc. IEEE, 121, no. 12, 1585 – 1588.

MatLab 6, 2000. User's Guide of the Fuzzy Logic Toolbox.

Takagi, T., and M. Sugeno, "Fuzzy Identification of Systems and Its Applications to Modeling and Control," *IEEE Trans. Syst. Man, Cybern*., SMC-15(1), 116-132 (1985).

Sugeno, M., & G. T. Kang, "Structure Identification of Fuzzy Models," *Fuzzy Sets Syst*., 28, 15 (1988).

Uhrig, R. E., Wrest, D. J., Hines, & J. W. August 1997, "Signal Validation Using An Adaptive Neural Fuzzy Inference System.", NUCLEAR TECHNOLOGY, VOL. 119.

Vo, T. V., B. W., F. A. Simonen, & S. R. Doctor 1990. "Development of Generic Inservice Inspection Guidance for Pressure Boundary Systems." *Nuclear Technology*, Volume 92(3), American Nuclear Society, La Grange Park, Illinois.

Vo, T. V., P. G. Heasler, S. R. Doctor, F. A. Simonen, & B. F. Gore 1991. “Estimate of Component Rupture Probabilities. Expert Judgment Elicitation.” *Nuclear Technology*, Volume 94 (1), AmericanNuclear Society, La Grange Park, Illinois.

Vo, T. V., F. A. Simonen, B. F. Gore, & J. V. Livingston. 1993. “Expert Judgment Elicitation on Component Rupture Probabilities for Five PWR Systems”. PVP-Vol 251. *Reliability and Risk in Pressure Vessels and Piping*. Pp. 115-140, American Society of Mechanical Engineers.

Vo, T. V., H. K. Phan, B. F. Gore, & F. A. Simonsen, S. R. Doctor February 1997, A Pilot Application of Risk-Informed Methods to Establish Inservice Inspection Priorities for Nuclear Components at Surry Unit 1 Nuclear Power Station, NUREG/CR-6181, Ver. 1, PNNL-9020, Ver. 1.

Wang, Li-Xin 1993, “ADAPTIVE FUZZY SYSTEMS AND CONTROL – DESIGN AND STABILITY ANALYSIS”, University of California at Berkeley. PTR Prentice Hall, Englewood Cliffs, New Jersey 07632.