Estimate the Failure Probability of Reactor Fuel Clad to Save Radiation Workers

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A nuclear reactor is a device used to initiate and control a nuclear chain reaction which main component is it's core. The cladding of the core of a nuclear reactor can be destructed or failed sometime. The probability of failure of a core systems can be measured by reliability based method. Such types of methods are FORM, SORM, Monte Carlo (MC) Method. Where method like MC is mainly used to check the accuracy of FORM and SORM. The purpose of this study is to determine the failure probability of reactor fuel clad. Then, on using Monte Carlo method to calculate uncertainty intervals and reduce the uncertainty level. TRIGA MARK-II reactor of Bangladesh has been taken as reference for the analysis. Our motive is also to compare the outcome with other methods as though it'll be clear which method is mostly reliable for the resolution. For the purpose of this thesis computer code EUREKA 2//RR has been used. This code is a revised version of EUREKA-2 which was originally developed for the reactivity accident analysis of nuclear power plants. To get a proper outcome there has also been used Response Surface method, Limit State condition and Safety Margin criteria and so on. This work may help to decrease the probability of failure in some various incidents like reactivity initiated accident (RIA), loss of coolant accident (LOCA), loss of coolant flow accident (LOFA) etc. The major contribution will be the improvement of failure probability of fuel clad and ensure the safety to radiation workers.
Keywords: TRIGA MARK-II, EUREKA-2, Failure probability, Reactor core, Radiation safety.

I) Introduction :

The primary purpose of research reactor is to provide a neutron source for research and their output (neutron beams) can have different uses depending on its level of neutron fluxes. . Although modern research reactors are designed with all safety features, however, various unusual events such as reactivity-initiated accident (RIA), loss of flow accident (LOFA), loss of coolant accident (LOCA) could be happened that could cause power excursion in the reactor core with leading reactor core to damage by release of large amount of fission products even though the heat transport system remains in perfect operating order. These serious accidental events needs to be addressed as a part of safety analysis and the results obtained from the analyses are compared with the pre-established results achieved from experiment or other computational simulation with the aim to verify whether the new core design fulfills the requirements of the safety constraints imposed in the design. In Bangladesh, we have 3 MW TRIGA Mark-II research reactor within the complex of Atomic Energy Research Establishment (AERE), Savar, Dhaka.Transient simulation addressed for TRIGA based on deterministic concept where inclusion of uncertainty or randomness in the inputs were not been taken into account during core modeling of TRIGA reactor. But to provide a more comprehensive and realistic measure of reactor safety, transient analysis of reactor should be interpreted probabilistically. This led us to estimate fuel failure probability as a response of fast transient simulation and techniques such as first order reliability methods (FORM), Response Surface Method (RSM), Monte Carlo Simulation (MC) and Directional Simulation (DS) methods have been applied in this context.

II) Aims of this work :

Specific Aims:

Because of various complexities and uncertainties, reactor thermal hydraulic safety should be viewed probabilistically and this is the approach taken in the present research. The research described in this thesis has been to investigate specifically the problems encountered in linking the technique of fast transient analyses with advanced methods of reliability analysis. Techniques such as first order reliability methods (FORM), second order reliability methods (SORM), and Monte Carlo simulation have been studied in detail in this context.The present methodology includes,

- Probabilistic descriptions of the variables related to limit state function of the problem.
- Developing a Response Surface Design (RSD) to generate an explicit limit state function and then determination of the reliability index and the probability of failure using FORM/SSORM algorithms
- Verify the estimated failure probability by crude Monte Carlo Simulation.

Figure 1 shows how these fuel pellets are affected due to RIA(Reactivity initiated accident). It is seen from the figure 1 that three types of fuel failure could be occurred due to RIA.

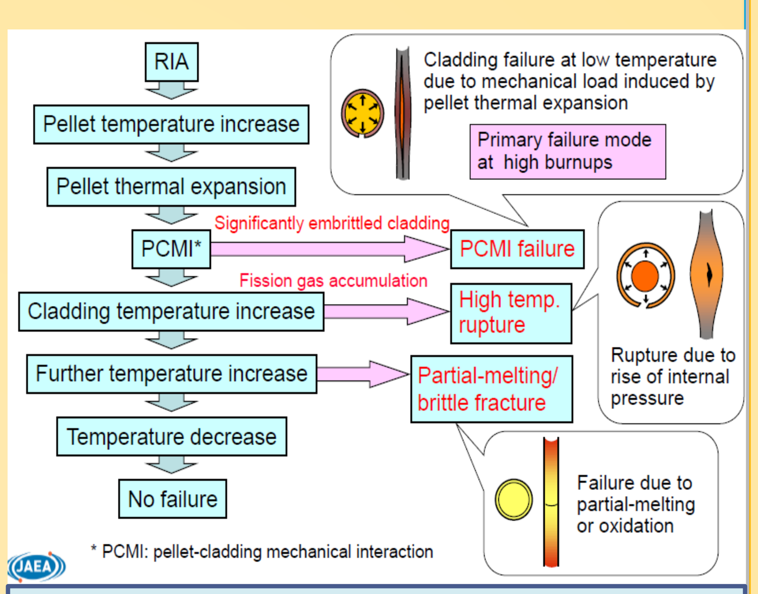

Figure 1- Illustration of impact of reactivity initiated transient analysis.

IV) Theory and algorithm:

The 3 mw triga mark-ii Research reactor is the only Reactor of bangladesh that has Been in operation over the last 26 years.The TRIGA core consists of 100 fuel elemets arranged in a Hexagonal array within the core Shroud as shown in figure 2.For higher power,forced convection cooling mode is required as seen in figure 3.Heat transfer mechanism in reactor from fuel clad to bulk coolant need to be understood prior to thermal hydraulic analysis. if the variation of heat flux in the heated surface is recorded against the surface water temp. Difference as in figure 4 ,the curve can be divided into four regions different codes like neutronic codes,structural codes ,Thermal Hydraulic codes has been used . EUREKA-2/RR has been used in this research work for reactivity accident analysis.

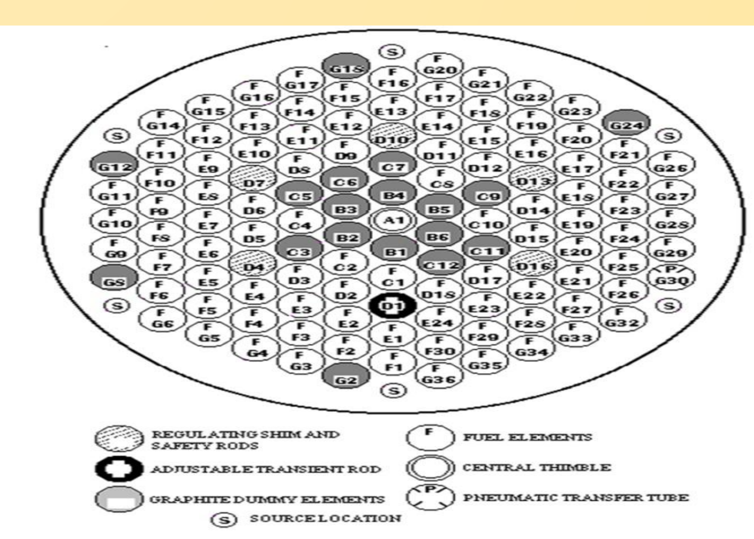

Figure 2-Configuration of 3 MW TRIGA Mark-II Research

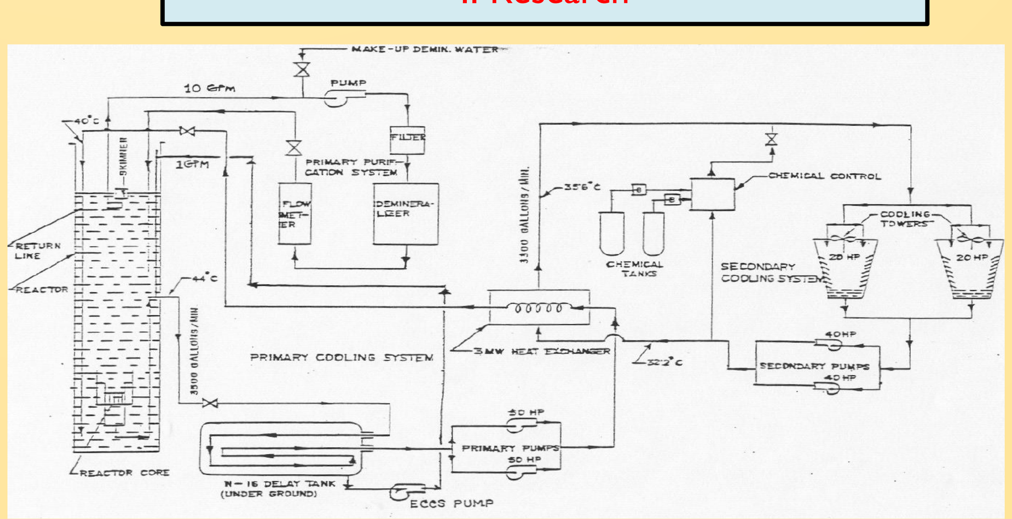

Figure 3 : Cooling System used in the 3 MW TRIGA Mark-II Research Reactor. [Courtesy-SAR (2001)

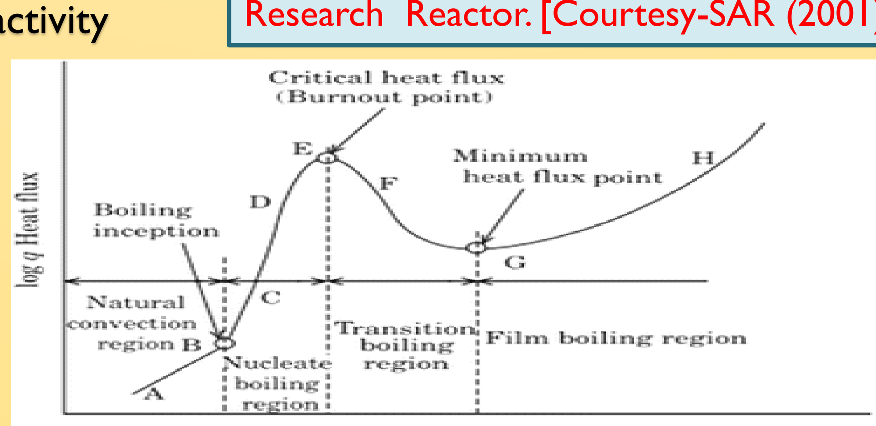

Figure 4- Heat transfer through different regions of boiling water

V) Results and discussions :

In order to apply the Crude Monte Carlo method in the present research, the following steps are required which for simplicity are discussed in terms of just two variables:

- Identification of limit state function which is expressed as follows for the present problem,
$$g(X) = T(X_2) - T(X_1)$$
- Transformation of the safty margin equation $g(\bar{X})=0$ into an equivalent safety margin equation $g(\bar{Z})=0$ in unit standard normal space(Z-space).
- Generation of random vectors in the Z-space based on the Box and Muller method.
- Transformation of the random variables into physical variables in order to perform transient simulation for evaluating the corresponding $T(\overline{X_1})$
- Computation of $g(\bar{Z})$ by subtracting $T(\overline{X_1})$ from the corresponding $T(\overline{X_2})$.

The above procedure is repeated for, say, a total of q trials and finally, P_f has been calculated.

Table-1:**Temperature distribution in the coolant in ^{o}C.**

Nodes	Channel-1	Channel-2	Channel-3	Channel-4	Channel-5
1	4.744E+01	4.671E+01	4.517E+01	4.367E+01	4.270E+01
2	4.664E+01	4.599E+01	4.461E+01	4.328E+01	4.241E+01
3	4.566E+01	4.511E+01	4.393E+01	4.279E+01	4.205E+01
4	4.462E+01	4.417E+01	4.321E+01	4.228E+01	4.167E+01
5	4.359E+01	4.324E+01	4.249E+01	4.177E+01	4.130E+01
6	4.268E+01	4.242E+01	4.186E+01	4.132E+01	4.097E+01
7	4.192E+01	4.173E+01	4.133E+01	4.095E+01	4.069E+01
8	4.124E+01	4.112E+01	4.086E+01	4.061E+01	4.044E+01
9	4.067E+01	4.060E+01	4.046E+01	4.033E+01	4.023E+01
10	4.020E+01	4.018E+01	4.014E+01	4.010E+01	4.006E+01

Table 2:**Pressure distributions in the coolant in kg/cm^2.**

Nodes	Channel-1	Channel-2	Channel-3	Channel-4	Channel-5
1	1.541E+00	1.542E+00	1.543E+00	1.544E+00	1.545E+00
2	1.537E+00	1.537E+00	1.538E+00	1.539E+00	1.540E+00
3	1.535E+00	1.535E+00	1.536E+00	1.537E+00	1.537E+00
4	1.533E+00	1.533E+00	1.534E+00	1.535E+00	1.535E+00
5	1.532E+00	1.532E+00	1.532E+00	1.533E+00	1.533E+00
6	1.530E+00	1.530E+00	1.531E+00	1.531E+00	1.531E+00
7	1.529E+00	1.529E+00	1.529E+00	1.529E+00	1.530E+00
8	1.527E+00	1.527E+00	1.528E+00	1.528E+00	1.528E+00
9	1.526E+00	1.526E+00	1.526E+00	1.526E+00	1.526E+00
10	1.523E+00	1.523E+00	1.523E+00	1.523E+00	1.523E+00

Table 3:**Power fractions for each heat slab.**

Slabs	Channel-1	Channel-2	Channel-3	Channel-4	Channel-5
1	1.683E-03	6.078E-03	1.287E-02	6.653E-02	2.4503E-03
2	2.163E-03	7.810E-03	1.653E-02	8.549E-02	3.148E-03
3	2.552E-03	9.211E-03	1.950E-02	1.008E-01	3.713E-03
4	2.475E-03	8.934E-03	1.891E-02	9.780E-02	3.601E-03
5	2.483E-03	8.963E-03	1.898E-02	9.812E-02	3.613E-03
6	1.916E-03	6.917E-03	1.464E-02	7.571E-02	2.788E-03
7	1.748E-03	6.309E-03	1.336E-02	6.907E-02	2.543E-03
8	1.519E-03	5.485E-03	1.161E-02	6.004E-02	2.211E-03
9	1.252E-03	4.520E-03	9.571E-03	4.948E-02	1.822E-03
10	9.954E-04	3.592E-03	7.607E-03	3.932E-02	1.448E-03

Table 4:Increase of P_f with the increase of trial numbers:

Trials No	P_f
50	1.6E-1
100	1.7E-1
500	1.72E-1
1000	1.75E-1

Discussion: The simulation provided clad temperature for which the safety margin value found zero which indicates that approximate response surface was fairly close to the true response surface. Based on this closeness of response surface with the true failure surface, reliability index with failure probability of fuel clad has been evaluated which is computed as ≈ **1.5390E-1.** Later, for more confirmation of the results obtained through FORM analysis, Monte Carlo simulation has been applied.

In Monte Carlo simulation, there were four experiments considered based on different sets of random numbers with 50, 100, 500 and 1000 trials. Taking into account $g(T_{design}, T_{evaluated}) = T_{design} - T_{evaluated}$ ($) ≤ 0, P_f has been computed in each experiment. **Table 4** provided estimated failure probability for all four experiments. It is noticed FORM results are relatively closer with the result of Monte Carlo simulation for smaller number of trials.

VI) Uncertainties :

The accuracy in reliability measurement depends on the analyst's present state of knowledge and experience about the structure, and the computed reliability is likely to change as more information becomes available. Lack of information about the uncertainties involved with different parameters may result in the prediction of structural reliability being inaccurate. There could be arise uncertainties like-

- **Physical uncertainties Model uncertainties Human error Statistical uncertainties**

VI) Conclusions :

Safety and security concerns about nuclear power reactors have played a significant role in hindering the expansion of nuclear power. While major accidents such as Three Mile Island, Chernobyl, and Fukushima have increased global awareness of safety concerns. However, nuclear power releases less radiation into the environment than any other major energy source. The purpose of the work is to minimize unwanted accidents in the reactor core. And assure safety for the workers,who works in a nuclear power plant.

VII) References :

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