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EMBRITTLEMENT AND ANNEALING

A POWERFUL METHODOLOGY FOR REACTOR VESSEL PRESSURIZED THERMAL SHOCK ANALYSIS

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ABSTRACT

The recent operating experience of the Pressurized Water Reactor (PWR) Industry has focused increasing attention on the issue of reactor vessel pressurized thermal shock (PTS). More specifically, the review of the old VVER-type of reactors (VVER 440/230) has indicated a sensitive behavior to neutron embrittlement. This led already to some remedial actions including safety injection water preheating or vessel annealing.

Such measures are usually taken based on the analysis of a selected number of conservative PTS events. Consideration of all postulated cooldown events would draw attention to the impact of operator action and control system effects on reactor vessel PTS.

Westinghouse has developed a methodology which couples event sequence analysis with probabilistic fracture mechanics analyses, to identify those events that are of primary concern for reactor vessel integrity. Operating experience is utilized to aid in defining the appropriate event sequences and event frequencies of occurrence for the evaluation.

Once the event sequences of concern are identified, detailed deterministic thermal-hydraulic and structural evaluations can be performed to determine the conditions required to minimize the extension of postulated flaws or enhance flaw arrest in the reactor vessel. The results of these analyses can then be used to better define further modifications in vessel and plant system design and to operating procedures.

The purpose of the present paper will be to describe this methodology and to show its benefits for decision making.

1. INTRODUCTION

Radiation embrittlement has been an important issue in the nuclear industry since the manufacture of the first commercial Pressurized Water Reactor (PWR) vessels in the late 1950's and early 1960's.

Reactor vessel Pressurized Thermal Shock (PTS) events are transients which result in a rapid cooldown in the primary system coincident with a high or increasing primary system pressure. The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The TMI event of 1979 was responsible for raising the concern of "pressurized thermal shock" to a high level of visibility in the 80's. TMI, along with an earlier cooldown event (during high system pressure) occurring at the Rancho Seco plant in 1978, together caused a realization that events resulting in sudden cooling of the vessel, while accompanied by high pressure, i.e., PTS, could actually occur.

Industry responses to the US NRC questions included thermal-hydraulic and fracture mechanics evaluations demonstrating that PWRs could continue safe operation, but concluded that some vessels may require additional measures to show vessel integrity throughout end-of-life. The evaluations were made solely in a deterministic basis using postulated design basis transients. It was decided that a more realistic evaluation should be done using statistical methods and probabilistic techniques.

The methodology that was developed by D.S. Ackerson et al. (1983) of Westinghouse, coupled probabilistic risk assessment (PRA) with thermal-hydraulic and probabilistic fracture mechanics (PFM) analysis to identify the dominating scenarios which lead to a PTS concern and to determine the risk of significant flaw extension from PTS transients.

This approach was used by Westinghouse subsequently by the US NRC to support their position and licensing requirements for PTS for all PWRs.

The present paper adapts this methodology that can be used to identify transients scenarios of concern and select the most adequate modifications that may be necessary to address PTS safety goals for a particular plant design, including the VVER reactors.

2. PROBABILISTIC PTS EVALUATION

The overall probabilistic PTS evaluation, illustrated on Figure 1, is divided into two basic parts. A broad risk assessment is performed that utilizes various lower resolution techniques to assess the PTS risk of literally thousands of hypothetical cooldown transients. Event tree analysis, stylized transient characterizations, and probabilistic fracture mechanics analyses are employed in order to estimate total plant PTS risk and to identify the specific transient scenarios that are anticipated to contribute most significantly to the total PTS risk. Next, a rigorous risk analysis is performed on each dominant transient scenario which is selected from the broad risk assessment.

Detailed thermal-hydraulic and deterministic fracture mechanics methods are employed to determine the precise risk of the dominating transients in relationship to PTS safety goals. If the PTS risk analysis shows unacceptable frequencies of vessel failure, an evaluation of modifications can be made to determine if an acceptable result can be achieved. A detailed discussion of the overall PTS evaluation is presented in the following paragraphs.

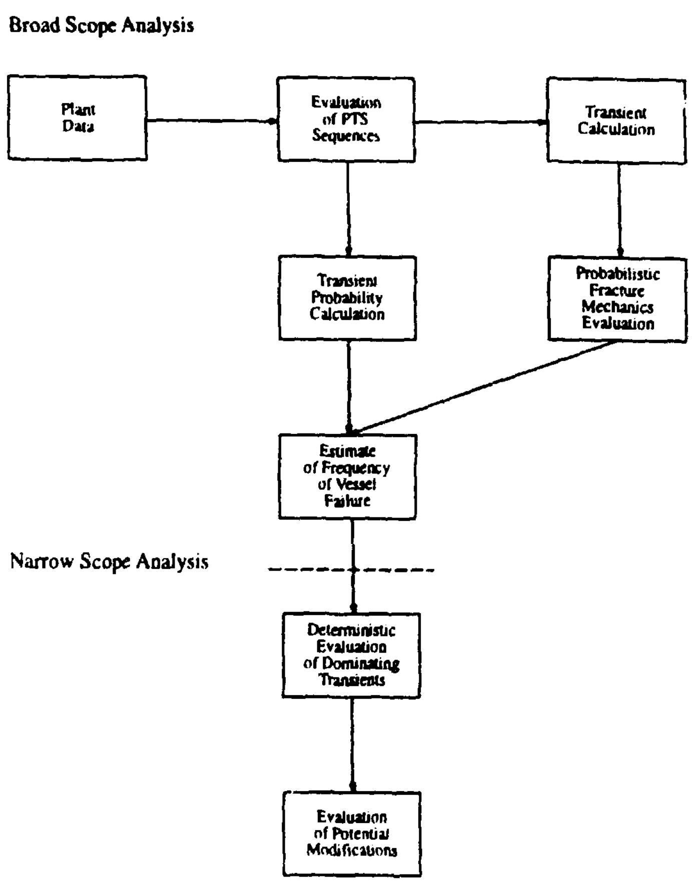

Figure 1

Methodology for Reactor Vessel Integrity Evaluation

2.1 PTS Risk Assessment (Wide Scope, Lower Resolution)

The first step in the PTS risk assessment is to identify, using event tree analysis, those sequences of events and associated frequencies that could potentially result in a pressurized thermal shock of the reactor vessel.
The frequencies depend upon both independent occurrences of these transients as initiating events and consequential occurrences from other events that are not by themselves cooldown transients. These other transients may be transformed into PTS transients as a result of plant system malfunctions or operator error.
The plant event tree, in response to being perturbed by each initiating event, models the probability of plant systems availability and malfunction for each possible sequence. Thus, the output of the plant model is a vector of cooldown states of varying degrees of severity and frequencies resulting from all possible sequences of plant response to all expected initiating events.

The plant cooldown state vector is then incorporated in the mitigation event tree that probabilistically models the effect of automatic or operator actions to terminate or lessen the effect of cooldown sequences. The overall result of the event tree approach is an end state vector of potential unmitigated PTS scenarios, grouped by common end state categories and decay heat levels, along with the associated frequencies.

The next step is to associate thermal-hydraulic characteristics with each cooldown transient end state. Because there can be literally thousands of end states and since only a small number prove to be of any practical concern, it is prudent to use simple approximations at this point. Using sensitivity studies and judgement based on experience, each end state is fit with a simple exponential curve, and a pressure is selected that is representative of the system pressure during the period of potential flaw extension.

The third step is to quantify the conditional probability of reactor vessel failure given that an exponential cooldown, which approximates the cooldown end state, occurs. The conditional probability curves are generated from probabilistic fracture mechanics (PFM) analyses using the Monte Carlo technique. A matrix of cases for given transient characteristics and inner surface RT_{NDT} values are generated to obtain results for generation of conditional probability curves.

For each case, a large number of deterministic fracture mechanics analysis trials (~ 10^6) are simulated using random values that are selected by a random generator from distributions defined for the pertinent input properties. The input properties, which have been treated as random variables, include: initial crack depth, initial RT_{NDT}, copper content, fluence, and the critical stress intensity values for flaw initiation and arrest. The probability of vessel failure for each case is determined by dividing the number of failures by the number of trials.

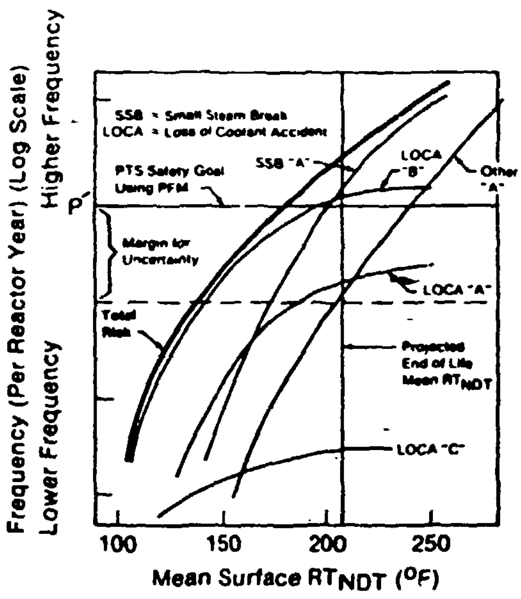

Figure 2
Plant Specific Frequency of Vessel Failure
Using Probabilistic Fracture Mechanics
(PFM) Analysis

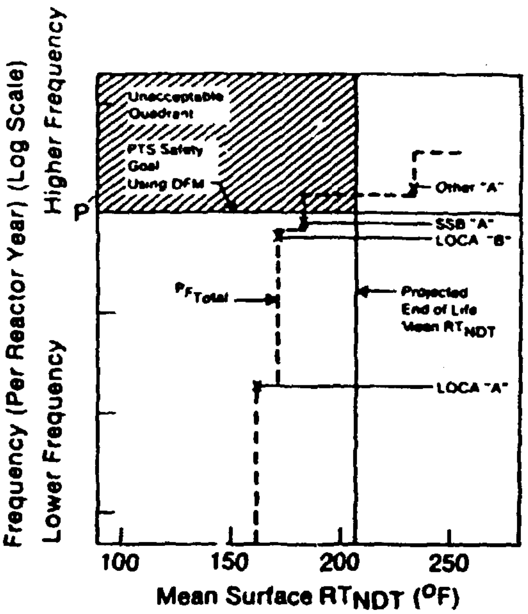

Figure 3
Plant Specific Frequency of Vessel Failure
Using Deterministic Fracture Mechanics
(DFM) Analyses

The final step is to associate a probability of vessel failure with each end state by multiplying the probability of the end state itself by the conditional probability of vessel failure given that the scenario occurs. If the probabilities of vessel failure that are associated with the end states are summed together, the total risk of vessel failure from PTS can be ascertained as a function of RT_{NDT}, and dominating transient scenarios can be identified. Figure 2 is a graph that shows total PTS risk as a function of plant RT_{NDT} values. Also shown is the risk associated with each cooldown scenario that is a significant contributor to the total PTS risk.

2.2 PTS Risk Analysis (Narrow Scope, High Resolution)

Once the probabilistic PTS risk assessment has been completed, the transient scenarios that dominate the plant total PTS risk can be selected for more detailed analysis. Referring to Figure 2, the scenarios that exceed the PTS safety goal in the RT_{NDT} range of interest are readily selected (LOCA "B" and SSB "A"). It is also appropriate to choose scenarios that fall below the safety goal within a band that accounts for uncertainty in the probabilistic analysis (LOCA "A" and Other "A"). The very low risk scenarios (LOCA "C" is an example), which incidently include many of the "design basis" transients that have been historically used for PTS decision making, are eliminated from further consideration.

Detailed thermal-hydraulic and deterministic fracture mechanics (DFM) analyses are then performed on the small group of selected transients. The results are then performed on the small group of selected transients. The results are plotted on a graph as shown in Figure 2. The abscissa of each "X" is the lowest RT_{NDT} at which vessel failure is predicted to occur for a specific transient. The ordinate is the frequency of occurrence of the transient scenario. A horizontal line is drawn to the right of each "X" because failure will always be predicted to occur at all RT_{NDT} values that are higher than the RT_{NDT} at "X". The total PTS risk curve (shown as a dashed line) is constructed by simply adding together the frequencies associated with each scenario that causes vessel failure at a given value of vessel surface RT_{NDT}. If the total PTS risk curve extends into the unacceptable quadrant, some modifications can be examined to determine if an acceptable result can be achieved. In Figure 3 the SSB "A" and/or LOCA "B" PTS transient(s) must be addressed to demonstrate acceptable PTS risk.

3. EVALUATION OF MODIFICATIONS

There are three general ways of affecting the PTS risk associated with a specific scenario. The frequency of the scenario can be reduced through the use of system or man-machine interface modifications (the "X" shifts downward). For example, installation of a block valve upstream of the secondary power operated relief valves might reduce the frequency of non-isolatable small steam breaks. The severity of the transient, if it occurs, can be reduced through the use of system modifications or man-machine interface improvements (the "X" shifts to the right). Heating of emergency core cooling water is an example of such a modification. Finally, methods such as flux reductions may be employed to reduce the vessel RT_{NDT} at the end of plant life (the vertical boundary of the unacceptable quadrant shifts to the left).

Potential actions that may be used, if necessary, to affect the risk due to PTS include; heating of emergency core cooling water, auxiliary feedwater flow limiting devices, PTS control and protection systems, flux reductions, vessel annealing, man-machine interface modifications, and specific equipment modifications. A cost-benefit analysis can be performed on individual modifications, or combinations thereof, that are found to affect PTS risk. In this way, the most cost effective method of achieving PTS goals can be selected.

4. CONCLUSIONS

A powerful methodology to address reactor vessel integrity in an integrated manner has been presented. This methodology uses probabilistic risk assessment techniques in combination with traditional deterministic methods. This allows a quantification of the risk of vessel failure and the identification of the dominating transient sequences.

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This approach is quite efficient to identify the modifications in vessel and plant system design that can better reduce the PTS risk.

This methodology can be applied to any PWR, including the VVER reactors which are quite sensitive to vessel embrittlement caused by the high content of impurities and high neutron flux in the beltline region.

5. REFERENCES

D.S. Ackerson, K.R. Balkey, T.A. Meyer, R.P. Ofstun, S.D. Rupprecht, D.R. Sharp, "A Quantitative Methodology for Reactor Vessel Pressurized Thermal Shock Decision Making", Nuclear Engineering and Design, 75 (1983), pp 405-414.