An Application of Probabilistic Fracture Mechanics to Reactor Pressure Vessels Including Multiple Initiation and Arrest Events

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SUMMARY

This paper describes the results which were obtained using probabilistic fracture mechanics in conjunction with Section XI, Appendix A of the ASME Boiler and Pressure Vessel Code. The objective is to assess the effect on the estimated probability of failing to meet integrity criteria for a nuclear pressure vessel beltline considering various important parameters such as copper content of the metal, number of effective full power years of plant operation, the initial RT_{NDT} and the RWST temperature. The accident conditions considered are the large loss of coolant accident (LOCA) and large steamline break (LSB).

The probabilistic methodology used for assessing the effect of the parameters mentioned above is based on the introduction of random variables into the deterministic methodology outlined in Section XI, Appendix A. In carrying out this probabilistic flaw evaluation analysis, the inputs are categorized either as constant values or random variables. For the random variables (copper content, phosphorous content, inherent K_{Ic} variation and inherent K_{Ia} variation), the form of the variability has been reflected by probability density functions. The fact that various inputs will be random variables means that the criterion variables, K_{Ic} and K_{Ia} are also random variables. In the present formulation, K_I values (the applied stress intensity factors) are given as a time dependent sequence of constant inputs. The probability of the assumed flaw initiating is estimated as the probability that $K_I \geq K_{Ic}$. The mechanics of the probability estimation procedure involve Monte Carlo computer simulation with importance sampling.

To estimate the probability of the assumed flaw initiating and not arresting within 75% of the wall thickness we estimate the probability of not arresting given initiation and multiply this by the probability of initiating (based on the definition of conditional probability). To estimate this probability we again make use of the simulation process but in addition we allow for multiple reinitiation and rearrest.

To carry out the sensitivity study, the above analysis is repeated for various settings of the parameter of interest. Among the four parameters studied, changes in copper content and number of effective full power years of service (i.e., fluence) have the greatest effect on the estimated failure probability.

1. Introduction

One scenario of potential accident conditions for reactor pressure vessels is the so called Class D conditions as defined in Section III of the ASME Boiler and Pressure Vessel Code (the ASME Code). Such conditions are not expected to occur during the life of a nuclear plant; nonetheless, extensive analyses are performed to assure the vessel's capability to accommodate[3] such potential accidents. The probabilistic analyses of this paper are structured as outlined in the summary.

2. Vessel Description

A typical commercial pressurized water reactor pressure vessel is a welded structure consisting of a cylindrical shell with top and bottom hemispherical heads, the top head being bolted in place. Water inlet and outlet nozzles are near the top of the vessel leading to the steam generator heat exchangers, pressurizer and pumps. Various other smaller nozzles exist. The diameters range from 11 to near 15 feet with thicknesses in the beltline from 6.5 to near 8.5 inches. Usually there are from two to four heat exchanger loops. The newer plants usually exhibit the higher design parameters. The vessel analyzed in this paper is for a three loop plant with dimensions typical of recent designs of commercial pressurized water reactor power plants.

3. Thermal/Hydraulic Stress and Fracture Mechanics Analyses

The LOCA and LSB are considered in this paper. During a double-ended, hot-leg reactor coolant pipe break (LOCA), the reactor vessel blows down very quickly followed by various rapidly occurring recovery events leading to safety injection pump flow from the refueling water storage tank (RWST) and finally recirculation to accommodate decay heat through the intact heat removal system.

For a large steamline break (LSB) accident, the reactor coolant temperature and pressure rapidly decrease. The safety injection pump delivers water somewhat as for the LOCA except a pressure buildup exists in the intact primary pressure system. The final circulation events are quite similar. Both the LOCA and LSB events subject the reactor vessel beltline to potentially significant thermal gradients, a typical example of which for a LOCA is sketched in Figure 1. Thermal stresses in the beltline result, an example of which is sketched in Figure 2 in the case of a LOCA as a function of time.

The postulated flaws are assumed to be semielliptical at the inside surface with a six-to-one aspect ratio. Stress intensity factors (K_I) are thus functions of flaw depth. A typical example of such a calculation for a LOCA is given in Figure 3.

For a deterministic evaluation, the remaining problem is to estimate the initiation fracture toughness (K_{Ic}) and arrest fracture toughness (K_{Ia}) for comparison with the applied stress intensity factors. If K_I exceeds K_{Ic} at a given time and crack depth, the crack, if present, is assumed to instantaneously initiate and propagate until K_I is less than K_{Ia} or the wall is completely breached. A direct evaluation is mitigated by the irradiation decay through the wall which in turn leads to varying toughness through the wall due to irradiation embrittlement.

A typical through wall irradiation variation is sketched in Figure 4. Taking irradiation embrittlement into account, K_{Ic} and K_{Ia} values can be plotted at a given time through the wall such as also sketched in Figure 3.

4. Failure Criteria

In a deterministic analysis, failure criteria must be established in order to assess whether or not failure occurs during the accident event. Such criteria must take a rational engineering approach. One typical approach is to assume that flaws greater than some agreed upon size (usually 0.5 or 1 inch) do not exist in the vessel wall when the event initiates, based on engineering judgement and pre- and in-service nondestructive examinations. If the analyses shows a smaller flaw initiates, then the flaw is assumed to be present and to propagate. Crack arrest, K_{Ia}, then governs. Should arrest occur, then both K_{Ic} and K_{Ia} govern, as appropriate, later in the event. The final flaw depth determines whether failure is postulated or not. For conservatism, final arrest at some fractional wall depth is usually taken as a failure criterion.

As an example, in Figure 3, a flaw would have to be almost 50% through the wall to initiate. If such a flaw were present and initiated, it would not be expected that the propagating crack would arrest prior to through wall penetration.

In the probabilistic study described in this paper, flaws of the stated size are assumed to exist with a certainty of one. For a given trial, as described later, which features potential multiple initiation and arrest events, failure is defined as initiating of the assumed flaw and nonarrest within 75% of the wall thickness.

5. Probabilistic Considerations

The probabilistic methodology used for assessing the integrity of the pressure vessel is based on the introduction of random variables into the deterministic ASME Code, Section XI, Appendix A. In carrying out the probabilistic flaw evaluation analysis, the inputs to the analysis are categorized either as constant values or random variables. For the random variables, the form of the variability will be reflected by probability density functions which are described later in this section. The fact that the various inputs will be random variables means that the criterion variables, K_{Ic} (fracture toughness) and K_{Ia} (arrest toughness) which are functions of these inputs, are also random variables. In the present formulation, K_I values (stress intensity) are used as a time dependent sequence of constant values. The probability of the assumed flaw initiating is estimated as the probability that $K_I \geq K_{Ic}$; for arrest, $K_I < K_{Ia}$. The various random variables and parameters are as follows:

PHOSPHORUS CONTENT - This is taken to be a random variable having a normal distribution with mean 0.012 and standard deviation of 0.001.

COPPER CONTENT - It is assumed that the variation in copper content can be described by a normal distribution with mean and standard deviation (0.001). The mean is varied to assess its affect on the probability of vessel failure. It should be noted that for a given fluence, the Cu and P contents determine the irradiation embrittlement.

FRACTURE TOUGHNESS - Using the K_{Ic} data available in the literature [1], a nonlinear regression computer program has been used to fit the following model:

$$K_{Ic} = a + b \tanh (T' - To)/C \qquad (1)$$

where K_{Ic} is the fracture toughness, T' is the adjusted temperature, and a, b, To and C are parameters to be estimated. The resulting model is

$$K_{Ic} = 159 + 116 \tanh [(T' - 73)/102] \qquad (2)$$

where the correlation between the observed K_{Ic} values and the corresponding predicted K_{Ic} values using the model is 0.82 and the standard error of estimate is s=11.

In order to represent this variation about the mean curve, we use a random variable KV, the inherent variation about K_{Ic}, having a normal distribution with mean zero and standard deviation of 11. Figure 5 shows this relationship schematically with the normal distribution about the curve at a given point.

ARREST TOUGHNESS - Using the K_{Ia} data available in the literature [1] in a similar manner, a mean arrest toughness curve has been obtained as

$$K_{Ia} = 119 + 107 \tanh [(T' - 109)/172] \qquad (3)$$

The correlation between the observed K_{Ia} values and the corresponding predicted values is 0.91 and the standard error of estimate is s' = 11. To represent this variation about the mean curve, we have used a random variable KA, the inherent K_{Ia} variation, having a normal distribution with mean zero and standard deviation of 11.

ADJUSTED TEMPERATURE - The adjusted temperature, T' used in the above expressions is defined as

$$T' = T - IRT_{NDT} - \Delta RT_{NDT} \qquad (4)$$

where T is the metal temperature which is input as a function of flaw depth and time into the transient, IRT_{NDT} is the initial reference nil ductility transition temperature and ΔRT_{NDT} is the reference temperature shift.

REFERENCE TEMPERATURE SHIFT - The reference temperature shift, reflecting the shift due to embrittlement, is calculated based on the expression in Regulatory Guide 1.99 [2] as

$$\Delta RT_{NDT} = \min \begin{cases} [40 + 1000 (Cu - 0.08) + 5000 (P - 0.008)] (F/10^{19})^{0.5} \\ 250 (1.67F/10^{19})^{0.2041} \end{cases}$$

CRITICAL FRACTIONAL DEPTH - It is assumed in this analysis that if a flaw has not arrested when it is three fourths of the way through the vessel, the vessel fails.

TRANSIENTS - Three transients are considered in the analysis. These are loss of coolant accident (LOCA) with the RWST of either 32 or 68^{o}F and large steam break (LSB).

MATERIALS - Both base metal and weld material have been compared in the analysis.

EFFECTIVE FULL POWER YEARS (EFPY) - A range of values of this parameter is used from as low as 4 years up to 32 years.

FLUENCE - The fluence is input as a series of depth dependent constants and is linearly related to the number of EFPY.

INITIAL RT_{NDT} - The initial reference nil-ductility transition temperature is a parameter for study to see how changing it affects the probability of failure.

6. Results and Conclusions

In this section we have summarized the results of the various parameter comparisons which have been run. The probability results were obtained under the conditions and assumptions described in Section 5.

It should be pointed out that we are using these probability estimates in a comparative way rather than an absolute way. That is, we are comparing the probability estimates obtained rather than suggesting that the estimated probabilities are exact.

The results in this section have been divided into 3 parts: a small flaw study, a large flaw study and a separate half-inch flaw depth.

SMALL FLAW STUDY - The initial flaw size was allowed to take on the value of one quarter inch, one half or one inch, in order to see how the change would affect the failure probability. As can be seen in Figure 6, the effect of flaw size was minimal but the effect due to changes in copper content was significant. It appears that the change in the initial RT_{NDT} has a minor effect. It should be noted that these results are for end of life only (32 EFPY). For less time (lower fluence) the probabilities of failure decrease rapidly. For example, the results for 4 EFPY all fall below 10^{-10}.

LARGE FLAW STUDY - The initial flaw was assigned a value 1, 2, 3, 4 and 5 inches and the probability results are plotted in Figure 7. The most obvious result is that although flaw size had no effect on probability of vessel failure for high copper content of 0.25, the flaw size had a marked effect for the low copper content of 0.05. The initial RT_{NDT} had no perceptible impact. Again for less fluence, the unreliabilities were seen to be very small early in plant life and increasing rapidly near one-half plant life.

HALF-INCH FLAW STUDY - In Figure 8 we are varying the copper content, the initial RT_{NDT} for weld metal and the number of effective full power years for a large steam break transient. As expected, the greater the copper content, the larger the probability of vessel failure. Also the number of effective full power years (fluence) and initial RT_{NDT} have the effect of increasing the probability of failure as they increase, hut to a lesser extent. In Figure 9 the transient of interest is the LOCA. The figures in the first column are for the weld metal and those in the second column are for the hase material. As before, the largest effect is due to the copper hut also notice that the probabilities are uniformly larger for the lower RWST. Again the initial RT_{NDT} and number of effective full power years have an effect hut to a lesser extent.

It would appear that copper content is the single most significant random variable (among those studied) affecting the probability of failure. Since in this study, copper content is a measure of irradiation sensitivity, this result is, in general, not surprising. It is to be recalled that in these studies, the unreliabilities are conditional probabilities, the most significant condition being the assumption that the transients occur. The probabilities of such transients occurring have been estimated at 10^{-4} or less per reactor year [3]. The assumption that the flaw is present and critically located introduces additional factors which decrease exponentially with increasing flaw depth. An overall estimated probability of failure is thus seen to be well below 10^{-6} per reactor year for all the cases examined.

References

1. Marston, Theodore U., ed., "Flaw Evaluation Procedures, Background and Application of ASME Section XI, Appendix A," EPRI-NP-719-SR, August 1978.
2. U.S. NRC Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," United States Nuclear Regulatory Commission, April, 1977.
3. United States Nuclear Regulatory Commission Policy Issue, "Pressurized Thermal Shock (PTS)", SECY-82-465, (November 23, 1982).

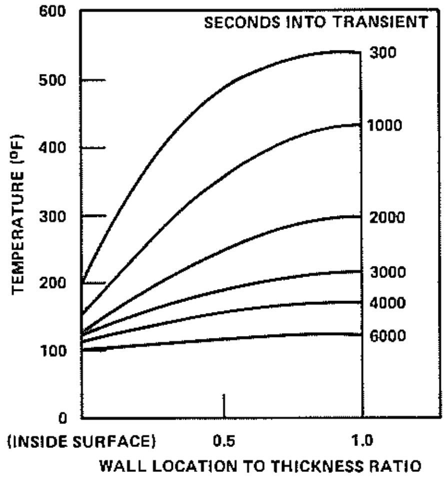

Figure 1 Temperature Distribution Through the Vessel Wall for a Large Loss-of-Coolant Accident

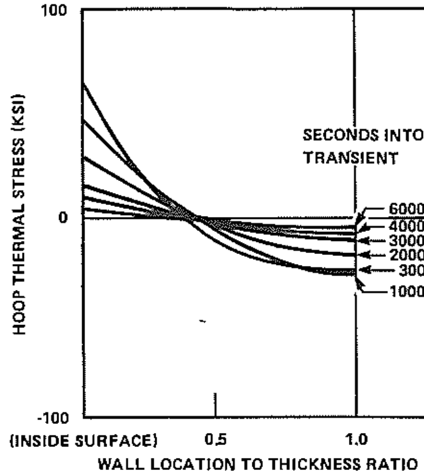

Figure 2 Thermal Hoop Stress Through the Vessel Wall for a Large Loss-of-Coolant Accident

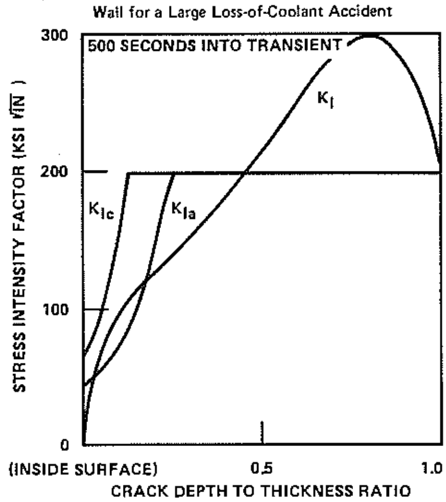

Figure 3 Stress Intensity Factors and Fracture Toughness for a Longitudinal Flaw 500 Seconds After a Large Loss-of-Coolant Accident (32 EFPY)

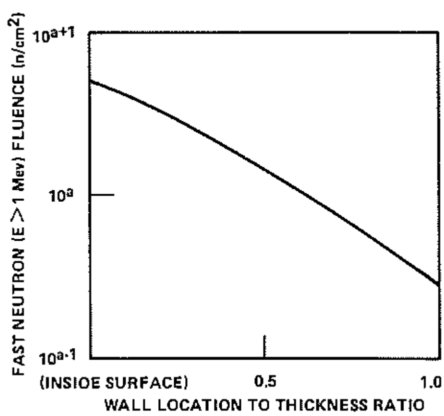

Figure 4 Radial Variation of Fluence in Vessel Wall

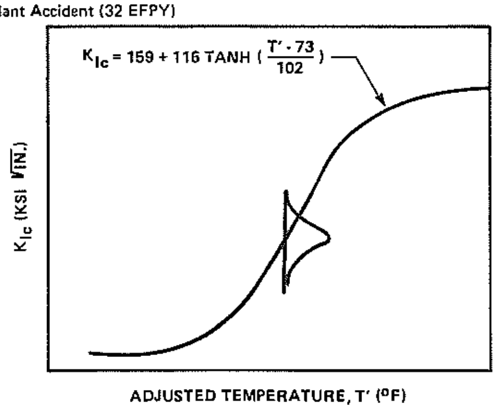

Figure 5 Fracture Toughness VS Temperature

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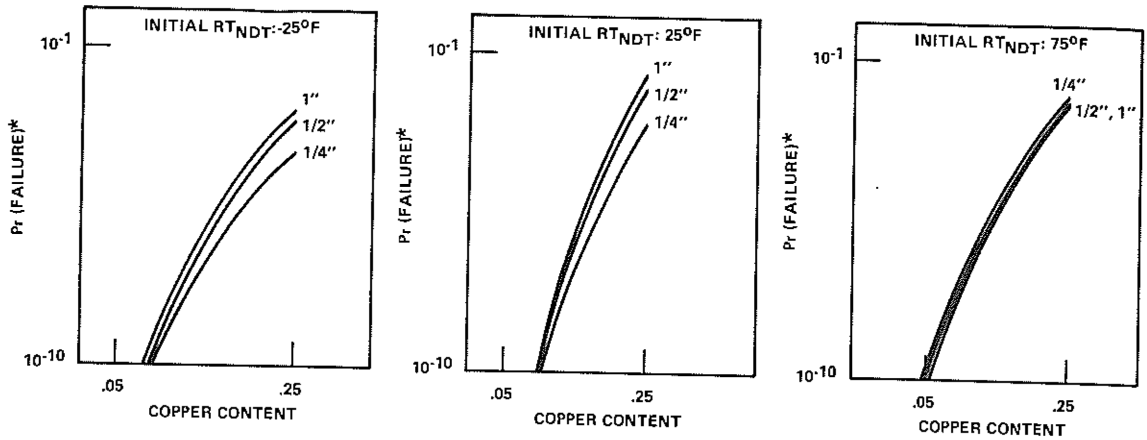

Figure 6 Small Flaw Study – The Transient is a LOCA for Base Metal with 32 EFPY

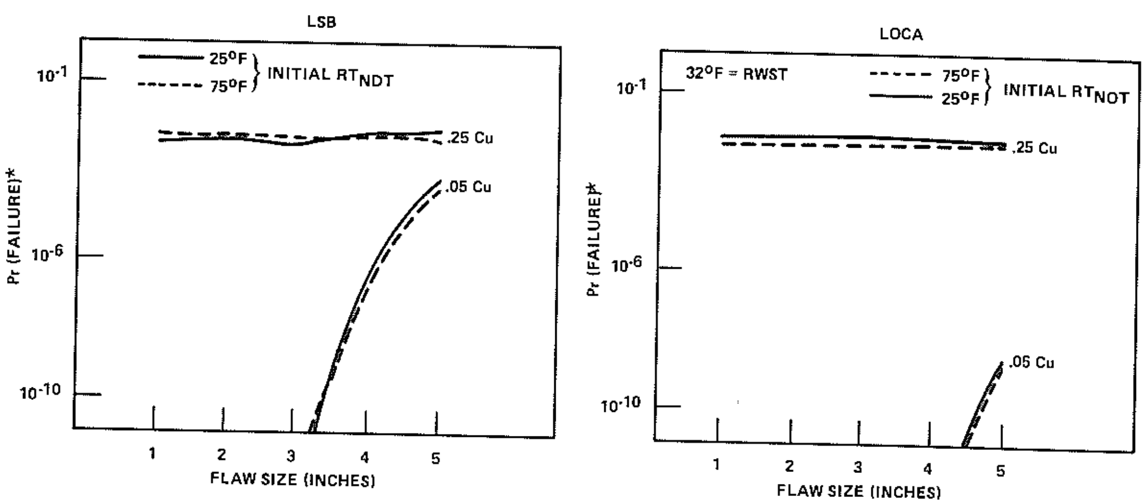

Figure 7 Large Flaw Study – for Base Metal and 32 EFPY

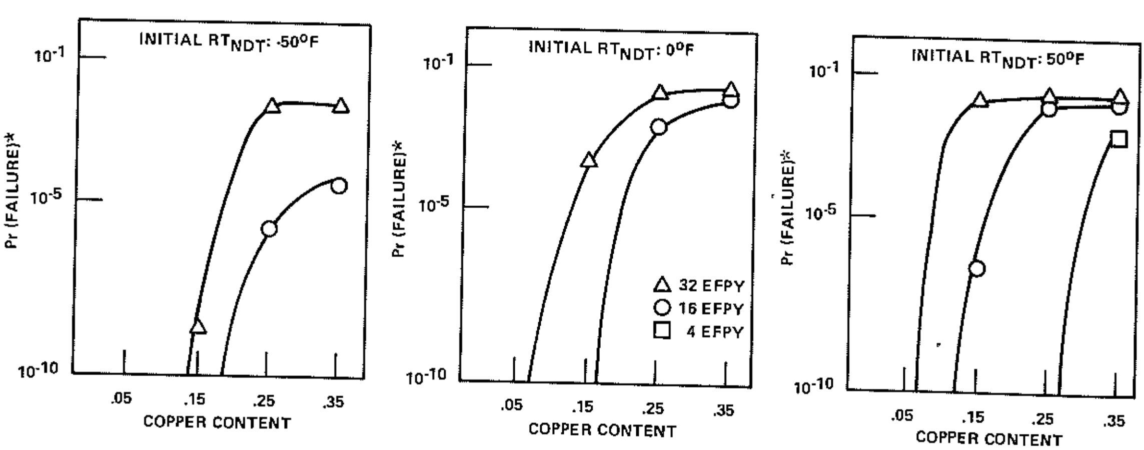

Figure 8 One-Half Inch Flaw Study – for Weld Metal Under LSB Conditions

*These are conditional probabilities. Actual failure estimates are well less than 10^{-6} per reactor year.

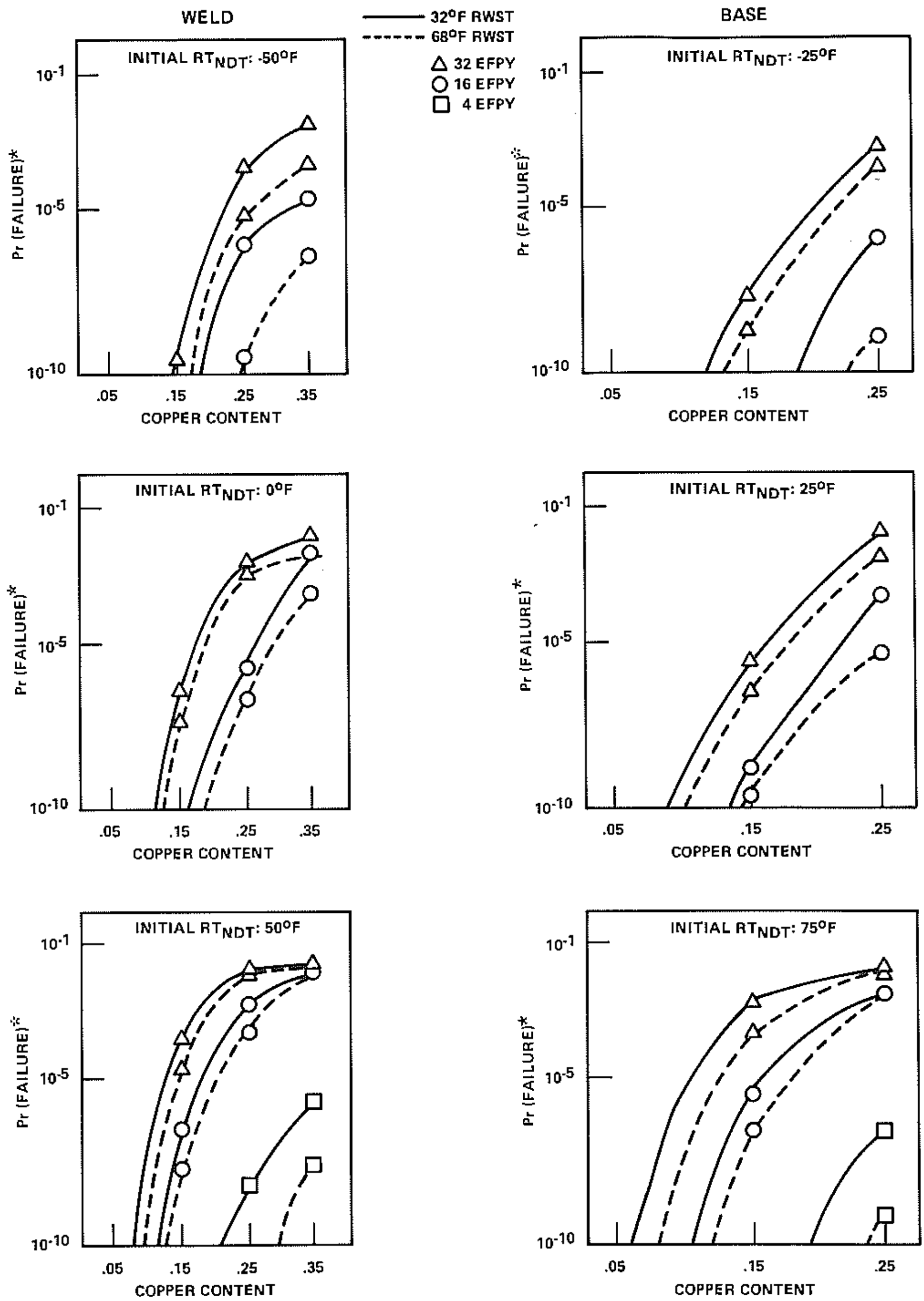

Figure 9 One-Half Inch Flaw Study for LOCA Transient

*These are conditional probabilities. Actual failure estimates are well less than 10^{-6} per reactor year.