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VERIFICATION OF REACTOR SAFETY CODES

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ABSTRACT

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The safety evaluation of nuclear power plants requires the investigation of a wide range of potential accidents that could be postulated to occur. Many of these accidents deal with phenomena that are outside the range of normal engineering experience. Because of the expense and difficulty of full scale tests covering the complete range of accident conditions, it is necessary to rely on complex computer codes to assess these accidents. The central role that computer codes play in safety analyses requires that the codes be verified, or tested, by comparing the code predictions with a wide range of experimental data chosen to span the physical phenomena expected under potential accident conditions. This paper discusses the plans of the Nuclear Regulatory Commission for verifying the reactor safety codes being developed by NRC to assess the safety of light water reactors and fast breeder reactors.

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I. INTRODUCTION

Over the years the Nuclear Regulatory Commission has evolved the safety philosophy of defense-in-depth. The first line of safety defense is to make sure the plants are designed and built correctly in the first place -- that is, to be sure the design, the materials, the fabrication methods, the construction practices and the testing and operation are of very high quality. The second line of safety defense is to provide protective systems to shut down the reactor plant in a safe condition in the event of equipment failures or breakdowns which can happen from time to time. To provide a third line of safety defense, the NRC staff postulates that serious accidents happen in spite of their very low probability, and then requires engineered safety features to mitigate the consequences of even these low probability accidents.

Thus, the safety evaluation of nuclear power plants requires the investigation of a wide range of potential accidents that could be postulated to occur. Many of these accidents deal with phenomena that are outside the range of normal engineering experience. For example, one of the severe accidents postulated by the NRC staff for a Pressurized Water Reactor (PWR) is the instantaneous double-ended rupture of the largest inlet cooling pipe leading to the reactor vessel. The staff requires that the plant design include a number of features, including emergency core cooling systems, to make sure that the consequences of such a loss-of-coolant accident (LOCA) are within acceptable limits. Similarly, one of the accidents postulated for a Fast Breeder Reactor (FBR) is a loss-of-coolant flow to the core coupled with a simultaneous failure of the reactor shutdown system. The staff requires suitable engineered safety features in the plant design to make sure that the consequences of this accident, termed a core disruptive accident (CDA), are within acceptable limits.

In order to assure that the plant safety features are adequate to mitigate these postulated accidents, the NRC sponsors a broad program of confirmatory safety research. Because of the expense and difficulty of conducting full scale tests covering the complete range of accident conditions that could be postulated, it is necessary to use sophisticated computer codes to evaluate these accidents. The central role that computer codes play in safety analyses requires that the codes be verified, or tested, by comparing the code predictions with a wide range of experimental data chosen to span the physical phenomena expected under accident conditions. The need to verify the quality and reliability of safety computer codes arises both from our desire to understand the safety margins in nuclear plants and from our desire to ensure public acceptance of the basis for making safety judgments.

In this paper the term code verification is synonymous with code assessment: to determine how well a code is capable of simulating the physical processes observed and measured in experiments; to assess the code's applicability to analysis of accidents in full scale plants; and to assess the uncertainty associated with the code's prediction of important safety parameters.

Further information on NRC's code development programs can be found in References 1-4, and on our code verification plans in References 5-7.

II. THE DEVELOPMENT, TESTING AND APPLICATION OF SAFETY CODES

NRC's plan for the development of advanced safety codes follows three phases as illustrated in Figure 1. The first phase, comprising the bulk of the effort, is to develop the overall framework and the detailed physical models embodied in the code. During the developmental phase, before the code is publicly released, many comparisons are made with the available test data, and models in the code are changed when necessary. Various sensitivity studies are performed to identify important parameters and in some instances to point out where additional research data may be needed. The emphasis during developmental checkout is on (a) numerics, (b) models of physical processes described by the constitutive equations, (c) system component behavior, and (d) integral system behavior. Once the code developers believe they have a satisfactory version of the code, it is frozen and all cases are rerun prior to documentation and public release of the code. The developers will then usually begin working on a newer version of the code which embodies more sophisticated models, improvements in numerics, improvements in user convenience, or inclusion of models applicable to different reactor systems (such as BWR vs. PWR for example).

The second phase of the code development process is an assessment of the code by an independent team. We believe that this step will help ensure an objective evaluation of the accuracy of the code, of the completeness of the code manual, and of the user convenience of the code. In this independent assessment phase, emphasis is placed on blind predictions of tests conducted in new test facilities and, also, significantly changed test conditions or system configurations in older test facilities. The purpose here is to exercise the code over a wider range of test conditions than was used in the development phase and thereby test the true predictive capability of the code. During this activity the results will be communicated to the code developers in order that follow-on versions of the code can correct any weaknesses found. The output of the independent assessment phase will be a code assessment report describing all of the comparisons of code predictions with test data, evaluation of code error, and extrapolation of that error to large scale plants.

When the NRC staff is satisfied that a version of the code has been adequately tested against experimental data and that the code meets NRC acceptance criteria, the code will be used in the application phase to aid the staff in assessing margins in the safety design of nuclear plants.

III. EVALUATION OF UNCERTAINTIES

One of the primary aims in developing advanced safety codes is to permit NRC to evaluate regulatory safety margins. To illustrate how this may be done, let us consider a comparison of a licensing evaluation model (EM) calculation with an advanced best estimate (BE) calculation of the peak cladding temperature (PCT) reached in a loss-of-coolant accident in an LWR. In order to make this comparison, one must know the uncertainty bands on the best estimate calculation. There are three sources of uncertainties that must be accounted for:

1. Code Model Errors (CODE ERROR)

   * physical models may be wrong or incomplete
   * numerical solution errors
   * stochastic phenomena not represented
   * etc.

2. Code Data or Built-In Coefficient Uncertainties (DATA)

   * fuel thermal conductivity
   * fuel-clad gap conductance
   * decay heat
   * heat transfer
   * momentum exchange
   * etc.

3. Reactor Plant Condition Uncertainties (REACTOR)

* availability of off-site power
* core conditions at time of accident
* functionability of safety systems
* etc.

One can begin by estimating the contribution of data uncertainties to the overall best estimate calculation uncertainty. This is done by assuming the code models are correct and considering fixed reactor conditions. Prior code sensitivity studies can tell us which parameters significantly influence the calculated peak clad temperature, while plots of basic test data on physical phenomena can tell us the uncertainty range of interest for each important parameter. One then carries out a large number of computer calculations in which the selected parameters are varied over the established range of uncertainty, using a prescribed sampling procedure. The result is a Response Surface -- a complex algebraic expression defining the effect of each parameter, x_i, on the peak clad temperature. Using the Response Surface in conjunction with Monte Carlo sampling from the individual probability distributions, $p(x_i)$ for each selected parameter, one finally obtains a probability distribution function for the peak clad temperature, $P_d(T)$ vs. T, which represents the uncertainty in the calculation due to code data uncertainties. Calculations of this type are already being carried out on a trial basis using older, less sophisticated codes.

The evaluation of uncertainty from code errors is less straightforward, and we have not yet worked out the complete formalism. During the independent assessment phase, a large number of comparisons between calculations and measurements will be made. If one plots the measured vs. calculated values of selected parameters from all comparisons, the result will be a series of scatter plots illustrated schematically in Figure 2. If the calculated values agree perfectly with the measurements, all of the points would fall on a 45° line. There will be a scatter of points, of course, and this scatter will be due to measurement uncertainty and code data uncertainties as well as code error. The next step is to strip out the measurement uncertainty and the code data uncertainties to get a statistical assessment of code error. The goal here is to obtain a probability distribution function of peak clad temperature, $P_c(T)$ vs. T, which represents the uncertainty in the calculation solely due to code errors.

In the next step, the code error probability distribution function $P_c(T)$ is convoluted with the code data uncertainty probability distribution function $P_d(T)$. The resulting probability distribution function $P_{cd}(T)$ represents the uncertainty in calculated peak clad temperature due both to code errors and to data uncertainties.

In assessing the safety margins for an actual reactor plant, one must also account for uncertainties in the reactor conditions before and during the accident. This uncertainty can be obtained by sampling from probability distributions representing the possible combination of reactor conditions, and, for each set of conditions, making best estimate calculation of the peak clad temperature. We know, however, that each calculated best estimated PCT has its own probability distribution due to code errors and data uncertainties. The resulting combination of code errors, code data uncertainties and reactor condition uncertainties is a probability surface represented by Figure 3.

Having obtained the PCT probability surface, one can obtain the probability of exceeding a given temperature T by integrating the volume under the surface intersected by the plane T = constant, and dividing by the total volume under the surface. The final result can be displayed as a probability distribution function as shown in Figure 4. An assessment of the margin of safety can be made by comparing the best estimate of peak clad temperature (BE) with the licensing evaluation model limit (EM) and calculating the probability of exceeding the licensing limit.

IV. CURRENT STATUS OF ADVANCED SAFETY CODES

The foregoing discussion has outlined NRC's plan for developing, testing and applying advanced safety codes. Although we have several safety codes under development, the principal advanced LWR system code is TRAC (Reference 3). An initial version has been developed and checked out, and an independent assessment is beginning, with the emphasis being on blind calculations of Semiscale-MOD 3, LOFT Nuclear Tests, PKL-Core 2, and LOBI. The final detailed uncertainty analysis and assessment of margins will not be concluded until all test data have been obtained from the planned LOCA/ECCS test program and the final version of TRAC has undergone independent assessment. This process will clearly take several years to complete.

With regard to advanced reactor safety codes, Figure 5 shows the codes NRC expects to use to analyze core disruptive accidents in fast breeder reactors. The principal CDA analysis code is SIMMER (Reference 4), and an initial version has been released and is undergoing checkout testing. Independent assessment of the advanced reactor safety codes has begun to a limited extent, but it will be many years before there is a data base adequate to permit assessment of regulatory margins in the same detail as for LWR's.

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PHASES OF CODE DEVELOPMENT, TESTING AND APPLICATION

FIGURE 1

CODE TESTING

Example: Measured PCT vs. Calculated PCT in SEMISCALE, LOBI, PKL and LOFT

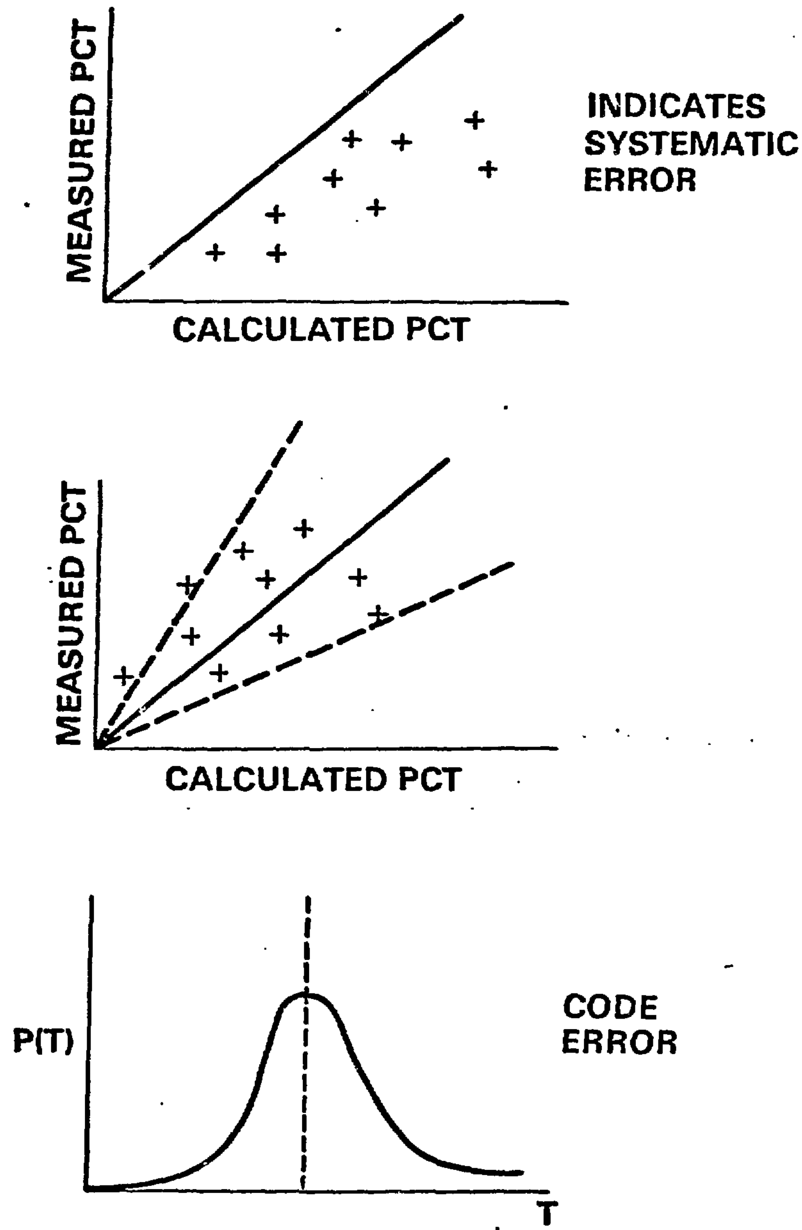

FIGURE 2

PEAK CLAD TEMPERATURE PROBABILITY SURFACE

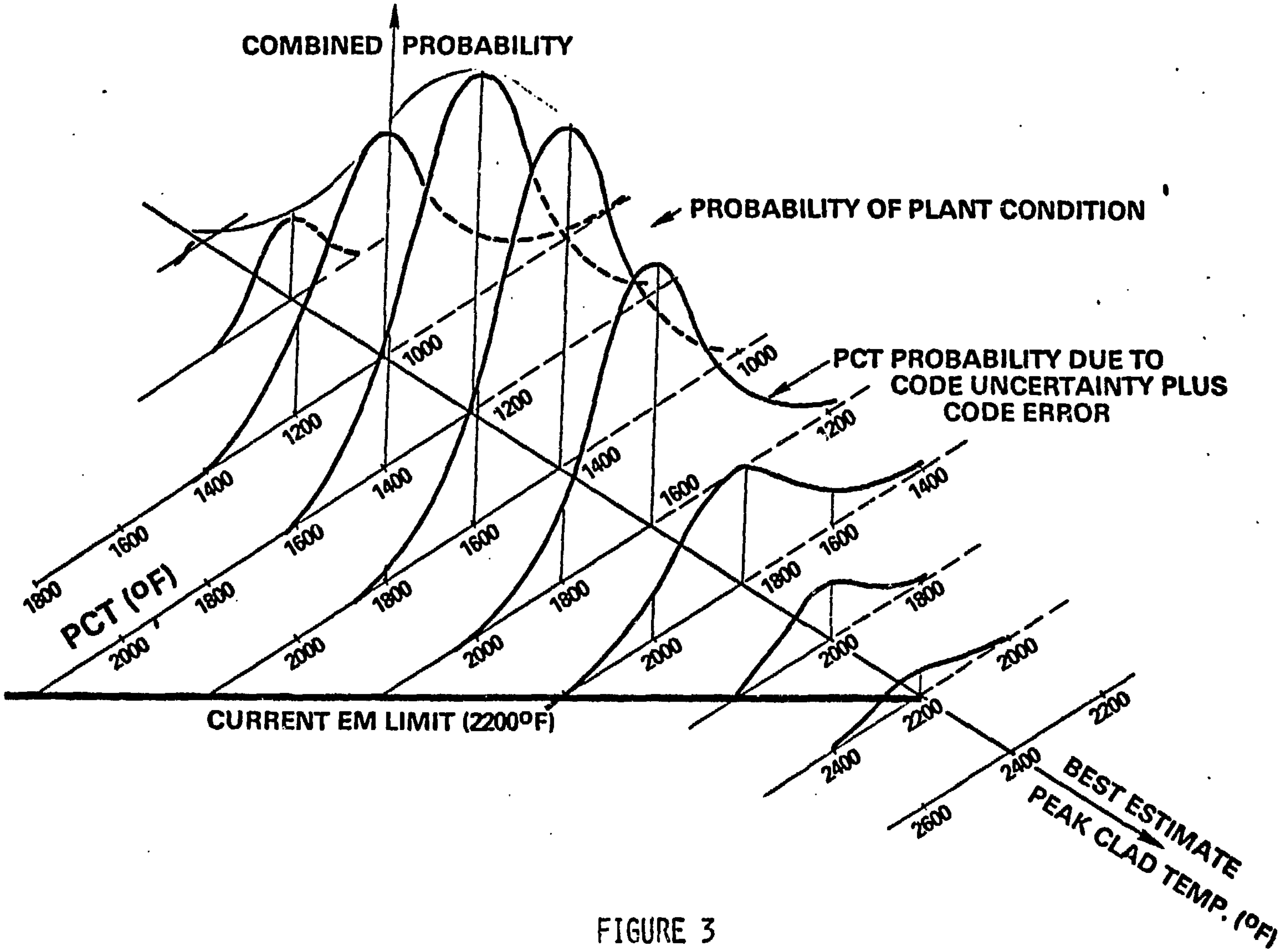

FIGURE 3

ESTIMATION OF MARGIN

- REACTOR CONDITION UNCERTAINTIES
- CODE MODEL UNCERTAINTIES
- CODE DATA UNCERTAINTIES

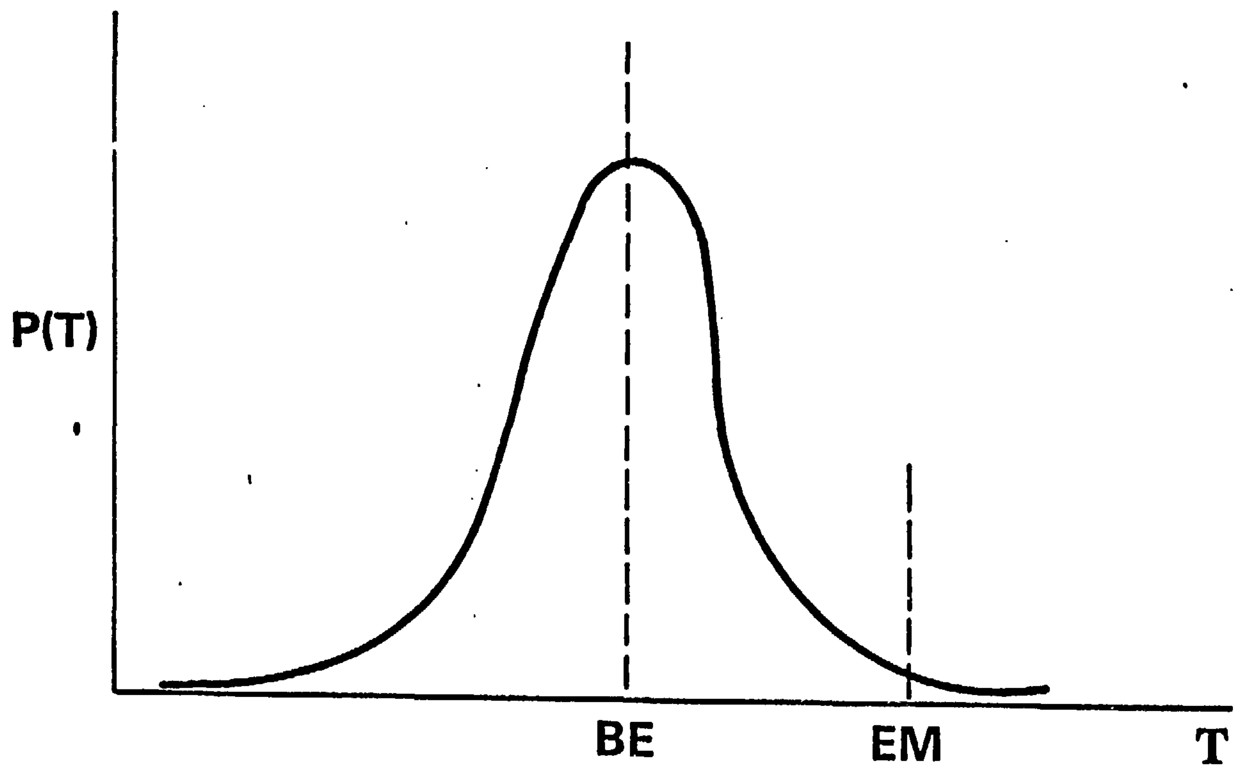

FIGURE 4

CORE DISRUPTIVE ACCIDENT ANALYSIS

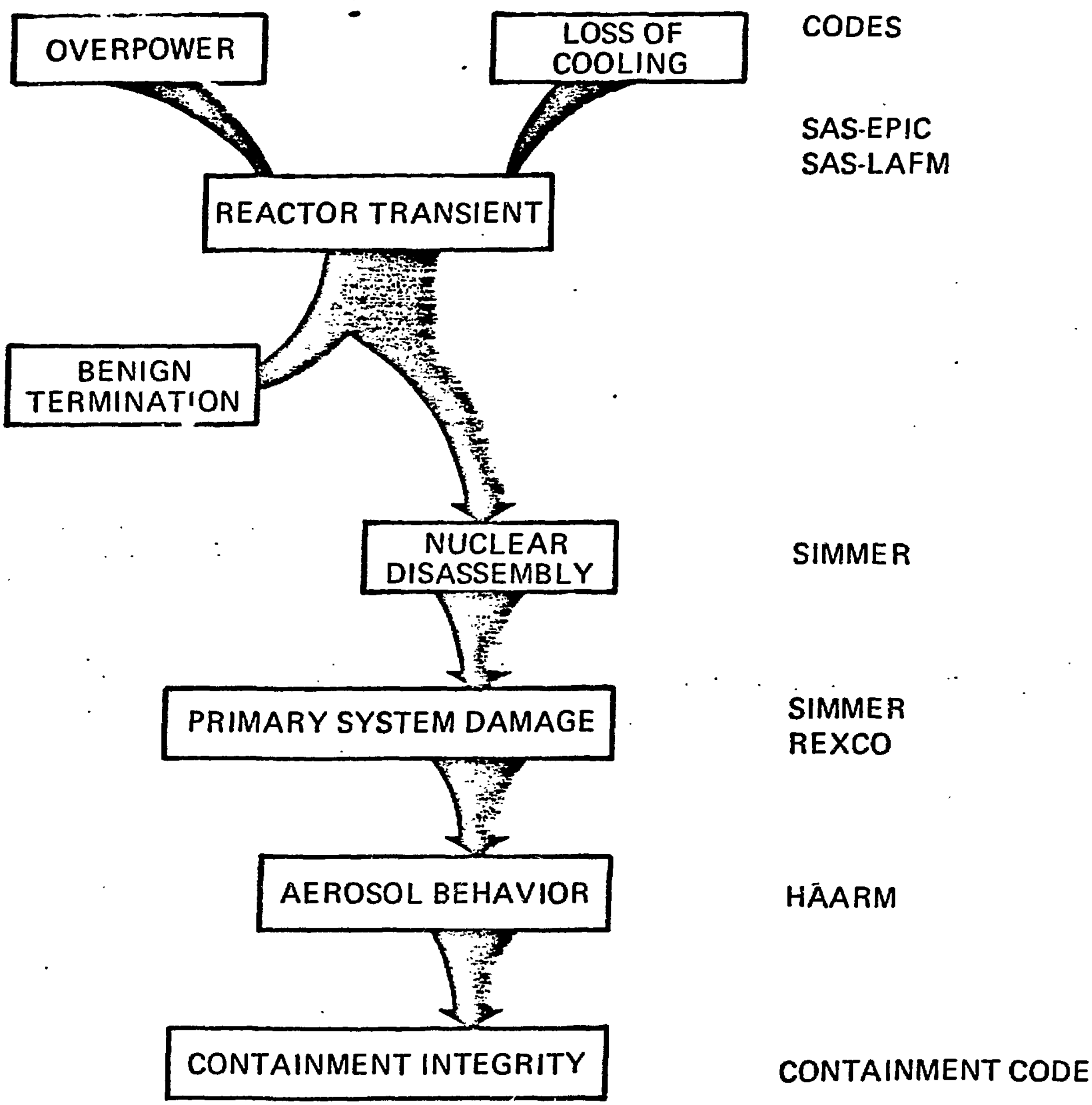

FIGURE 5