LOCA ANALYSIS METHODOLOGY

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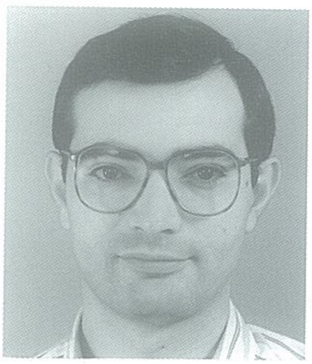

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INTRODUCTION

In order to ensure core cooling in the event of loss of cooling accidents, commercial reactors should be provided with Emergency Core Cooling Systems (ECCS). Standard 10CFR50.46, published in 1974, requires analyses of postulated LOCAS accidents and stipulates the acceptance criteria for demonstrating that the ECCS systems are capable of maintaining core integrity and cooling. Appendix K of this standard indicates the approved calculation methods for performing the analyses. The acceptance criteria and calculation methods were established in accordance with knowhow and state-of-the-art technology at that time, both as regards available experimental data and the capacity and development of existing calculation programs. As a result, 10CFR50.46 and Appendix K thereto display a considerable degree of conservatism that made up for the uncertainties of the time.

In the decade following publication of 10CFR50.46, an enormous experimental research effort was carried out (separated and integral effects) in parallel with development of best estimate codes, and thus we currently have a better knowledge of the physical phenomena that occur during this type of accident and proven, realistic calculation tools. There is evidence that there are wide margins between the results obtained with realistic and conservative methods that can be unnecessarily detrimental to operation, above all in the new advanced fuel designs, and on the other hand the presence of conservative factors can result in a prediction of accident evolution that strays from reality.

As a result of the above, the NRC has revised 10CFR50.46 (effective since November 1988). While this revision maintains the acceptance criteria, it permits the use of realistic calculations, provided that the calculation uncertainties are quantified and included when comparing the results with admissible values.

PROJECT EXPOSITION

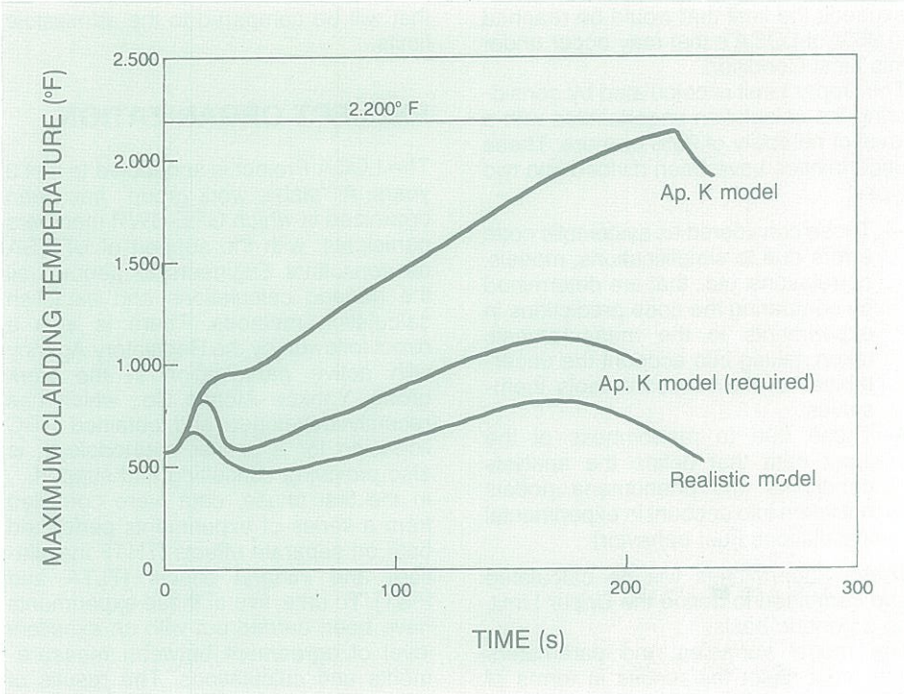
Taking into account the new regulatory tendencies, the availability of advanced calculation tools, and the anticipated benefits in plant safety and operation, the Spanish BWR Power Plant Owners Group (GPE-BWR) has decided to develop a "Methodology for the Analysis of Loss of Coolant Accidents in BWR Reactors" from the standpoint of Emergency Core Cooling System operation verification in accordance with the new revision of standard 10CFR50.46. The Project scope includes methodology licensing from the Regulatory Agency, which will provide for independent performance of this type of calculation and a better knowledge and use of the real existing margins with regard to the established limits.

The basic elements of the Project components are as follows:

— calculation tools
— application methodology.

The calculation tools form the mathematical framework that serves to calculate the system's evolution and response to

the postulated accident. For this purpose, the thermohydraulic and fuel codes that best provide for simulation of the physical phenomena that occur in BWR reactors during this type of transient have been selected.
Code TRAC-BF1, which was developed in a joint effort by the NRC, EPRI and General Electric and contains specific features and models for BWR reactors, has been selected as the most comprehensive and suitable calculation tool for LOCA analysis.
The postulated scenario assumes large and small breales in the Nuclear Steam Supply System piping in BWR/3 and BWR/6 reactors. On one hand, the physical phenomena that occur during the different phases of the postulated accident have been reviewed, and on the other, the suitability of the calculation and simulation capacities of code TRAC-BF1 for LOCA analysis and calculation of maximum temperature evolution in the fuel element cladding (PCT) have been assessed.
TRAC-BF1 includes the necessary models to realistically predict the phenomena observed in experimental installations and is considered (RG 1.157) as a realistic calculation tool. This code is available by virtue of the participation of GPE-BWR members in the ICAP Program, for which models of the corresponding plants have been developed.



F I *Tipical evolution of PCT*

BASES FOR ESTABLISHING A METHODOLOGY

10CRFR50.46 requires analysis of the possible loss of coolant accidents by considering the entire range of possible ruptures insofar as size and location are concerned, as well as the single most restrictive failure in the ECCS.
The acceptance criteria for ECCS operation are also established, in terms of Peak Cladding Temperature (PCT) calculated for the fuel element hundles (PCT) (should not exceed 2200 F) and other limits for maximum oxidation and hydrogen generation due to reaction with steam, as well as long-term maintenance of core cooling.
The most restrictive criterion is the one regarding maximum cladding temperature, as the others are usually satisfied if this one is fulfilled. The recent revision of 10CFR50.46 prescribes two possible ways to calculate this:
On one hand, former Appendix K, which remains in force, establishes certain conservative characteristics for the "Evaluation Models", some of which are excessively conservative and thus unnecessarily detrimental to the results such as those regarding:

— heat sources, as regards initial power and heat decay levels;
— heat transfer after CHF (critical heat flux), without giving credit to the return to nucleate boiling.

By considering that the requirements of Appendix K could be divided between required and acceptable, it was possible to establish analysis methodologies by using less conservative alternatives for the latter. While still in compliance with the Appendix, these methodologies provided results with a wide margin with regard to the admissible limits. Acceptance of this alternative is included in document SECY-83-472.
On the other hand, the new standard revision permits the use of realistic models with sufficiently contrasted calculation codes, provided that the calculation errors and uncertainties are quantified and included in the final results for comparison with the acceptance criteria.
Thus, there are currently two possible LOCA analysis alternatives that allow for reducing calculated PCT values and increasing the existing margins to accommodate possible design modifications, relaxing of technical specifications or to increase flexibility of operation.
The first one would consist in using an "evaluation model" that includes conservative elements only for the characteristics "required" by Appendix K, with properly justified realistic options for the 'acceptable" characteristics. Since the available calculation codes have been developed with realistic (best estimate) models and correlations, this alternative would involve modifying these codes so that they use the models "required" by Appendix K.
The second alternative would be based on the use of properly validated and contrasted realistic codes with plant models under the most probable rated conditions, and on analysis of all possible errors and uncertainties due both to the code and to the data that define the model, in order to definitively determine the degree of error in the final results (PCT) and ensure that the value to be compared to the admissible limits contains a degree of conservatism consistent with the calculation uncertainties (typically a probability level of 95%).
Figure 1 shows PCT evolution for the Design Base Accident (guillotine rupture ot the Recirculation suction line) with typical results obtained by using different calculation models.

GPE-BWR METHODOLOGY

The methodology designed by GPE-BWR is along the lines of realistic calculations ans has the operating advantages implied in using an evaluation model for practical and systematic application in safety and licensing calculations.
10CFR50.46 requires PCT calculation for the Limit Condition, which is defined as the worst-case break insofar as location, size and single most unfavorable failure are concerned.
This Limit Condition will be determined by analyzing the entire break spectrum with the Realistic Model: realistic hypotheses and rated plant conditions.
Once the Limit Condition is defined, the Upper Limit of the PCT must be calculated under these conditions, i.e. that which

exceeds the limit that would be reached in 95% of LOCA's that may occur under this Limit Condition.
The Upper Limit is calculated by considering the calculation uncertainties with a level of reliability of 95% or more. These uncertainties have been divided into two types:

— Those considered as systematic code errors due to simplifications, models, correlations, etc., that are determined by comparing the code predictions in experiments to the measurements taken, taking into account the uncertainties of the measurements themselves;
— Those due to randomness of the input data that define the analysis conditions and phenomena/models not taken into account in experimental installations (fuel behavior).

These uncertainties will be calculated and combined to define the Upper Limit, on a generic basis.
The model variables and parameters that most affect the results in terms of PCT will be determined on the basis of sensitivity analysis. This will serve to define an "Evaluation Model" that insures that the PCT value obtained therewith will be above the Upper Limit. The Evaluation Model results are the ones that will be compared to the admissible limits.

PROJECT ORGANIZATION

The LOCA Project is scheduled to last 3 years. A "stable work group" has been organized in which GPE-BWR members participate, with the support of UITESA as consultant Engineers to perform all the generic calculations and establish calculation methods. There is also a direct followup by the Regulatory Agency with active participation in the work group. Yankee Atomic Co., which has recently developed and obtained NRC licensing for a similar methodology, is also providing consulting and support.
In the first phase, data were compiled from a series of experiments performed both on separate effects (THTF installation) and integral effects (TLTA and FIST). To date, five of these experiments have been carried out with an excellent level of agreement between measurements and calculations. The results of these calculations have led to conclusions regarding modelling criteria and more suitable correlations that will be applied to the plant models. Once the experiment simulation is completed, the plant models developed within the scope of the ICAP Project will be adapted, and the data base for auxiliary fuel analysis codes (FRAP-T6) will be established.
In the second plase, the basic calculations were programmed with realistic plant models that encompass the entire break spectrum, for subsequent categorization of the parameters that have the greatest influence on the PCT, which will be the basis of the sensitivity analysis. In addition, the systematic code errors were quantified by comparing calculations and experiments and considering the possible uncertainties due to instrumentation.
Finally, in the third Project phase, sensitivity analyses will be performed that will be used to select the parameters that will serve, once their probabilistic distribution is established, to delimit the random calculation uncertainties on one hand and on the other to define the Evaluation Model. In this phase, all the generic calculations needed to justify the reduction of the break spectrum and to satisfy other limits in the practical imprementation of the Methodology will be performed. These will be included in the corresponding Application Procedures.
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