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PROBABILISTIC WHOLE CORE DAMAGE ANALYSIS USING THE SSYST FUEL BEHAVIOUR CODE

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Since 1975 methods have been developed to assess more accurately the possible extents of fuel rod damage in a PWR during a LOCA, including the associated probabilities /1/. The procedure starts out from a detailed probabilistic treatment of the single fuel rod, extends to rod bundles, and finally to the whole core. The radial power distribution is accounted for by introducing rod classes of similar power.

As an important tool the code RELAP4 is used for (a) the analysis of the primary system and (b) a set of single channel analyses, both for the blowdown phase only. Coolant pressure, coolant temperature and heat transfer coefficients for a number of axial nodes are communicated as boundary conditions to the code system SSYST-2, which calculates the transient behaviour of single fuel rods during the complete LOCA. The best-estimate code system SSYST is highly modular, with a control language of its own /2/. Separate modules compute transient temperature fields, gap conductance, internal gas pressure, cladding deformation etc. Physical modelling and algorithms in SSYST have been optimized for fast execution, leading to typical ratios of 1:15 for the machine times of an SSYST LOCA analysis and the associated RELAP4-runs, respectively. Fig. 1 shows as a typical output from an SSYST LOCA analysis the transient cladding temperature and circumferential strain of a PWR fuel rod (under extreme conditions).

The probabilistic analysis /3/ starts with sampling input data for the RELAP4 and SSYST codes from a parameter space containing (a) more global parameters (e.g. pump characteristics, initial power) and (b) more local parameters, which may vary from one rod to the next (e.g. material properties, gap conductance, pellet and cladding geometries). Considering the high computing costs involved, global parameters needed as input to RELAP4 have, so far, been varied only in the vicinity of a supposedly worst case set. The much larger group of input parameters for the transient fuel rod performance code SSYST has been varied freely. In toto, the probabilistic analysis relies by now on a data base of 3 primary system analysis runs and $3 \times 20 = 60$ single channel analysis runs of RELAP4, and more than 100 SSYST runs.

Output of these LOCA simulations is processed by a stepwise multiple linear regression scheme (with some 30 ansatz functions) to obtain for each relevant output quantity one response surface equation. This is used in later steps as a cheap substitute for the expensive original codes. In combination with the regression analysis employed, the Latin Hypercube sampling scheme seems to have sizeable advantages in randomly sampling from the space of input parameters. The quality of the response surfaces obtained is quite satisfactory, judging by statistical goodness of fit criteria and by checking the physical significance of the dominating terms.

Starting from known (or reasonably assumed) probability density functions (PDF) one obtains easily, by means of the corresponding response surface equations and standard Monte Carlo procedures, PDFs for the output parameters of interest. Main results obtained so far are as follows:

For the peak cladding temperatures (PCT) rather reliable PDFs can be obtained (Fig. 2). They show quite clearly the very low probability of reaching PCT > 1200 K.

The PDFs obtained for the final circumferential cladding strain ε (Fig. 3) have not quite the same degree of confidence. This is explained by the strain rate's $\dot{\varepsilon}$ strong and non-linear dependence on the cladding temperature, which in this respect is a stochastic variable.

Still, the PDFs for the strain ε seem to be of much greater concern. At least for the highest-loaded rod classes, maximum strains $\varepsilon > 30$ pct. cannot be excluded. The dependence on local rod power is high. Maximum strains show pronounced axial bunching, with a width of some 20 cms, when conservatively the initial high axial peaking factor is taken as independent of burnup. This raises the question, whether significant reductions of coolant channel areas may impede emergency cooling in some elements, especially if clustering of reduced coolant channels is possible.

Starting from the single rod PDFs for final strain, the probabilities for the numbers of significantly strained rods and reduced coolant channels (Fig. 4, 5) have been computed by Monte Carlo techniques, along with possible cluster patterns. As an aid for identifying those patterns which deserve a more detailed study, an equivalent cluster diameter has been tentatively defined. The PDFs evaluated for this diameter seem to indicate that even for the highest-loaded fuel elements diameters greater than 3 pitch lengths can be practically excluded. This means that emergency cooling is not impeded significantly.

The probabilistic treatment of fuel rod damage during a LOCA seems to predict rather mild consequences, as far as emergency cooling is concerned. This statement is based on the available best-estimate codes, the assumed conservatism in the LOCA scenario, and extrapolations from the so far treated highest-loaded fuel element to the whole core.

References:

/1/ Contributions to the Semiannual Reports of the Projekt Nukleare Sicherheit, starting from Report KfK-2375, Karlsruhe (1976)

/2/ W. Gulden (comp.) Dokumentation SSYST-1, Report KfK-2496, Karlsruhe (1977)

/3/ W. Sengpiel, Ph. D. dissertation, Karlsruhe (1979)

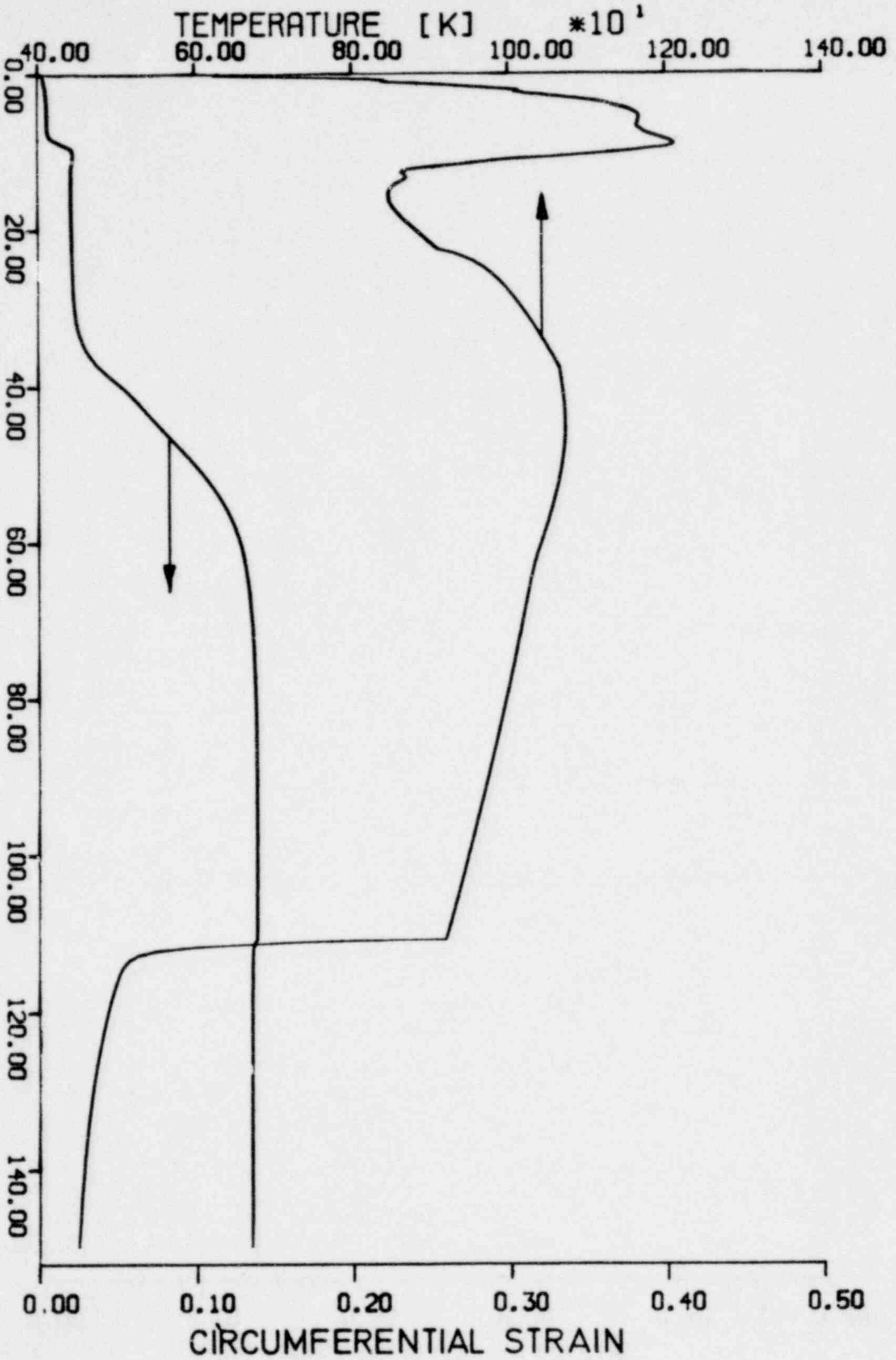

Fig. 1: Transient cladding temperature and circumferential cladding strain respectively for a PWR fuel rod following a postulated 2F-LOCA evaluated by SSYST

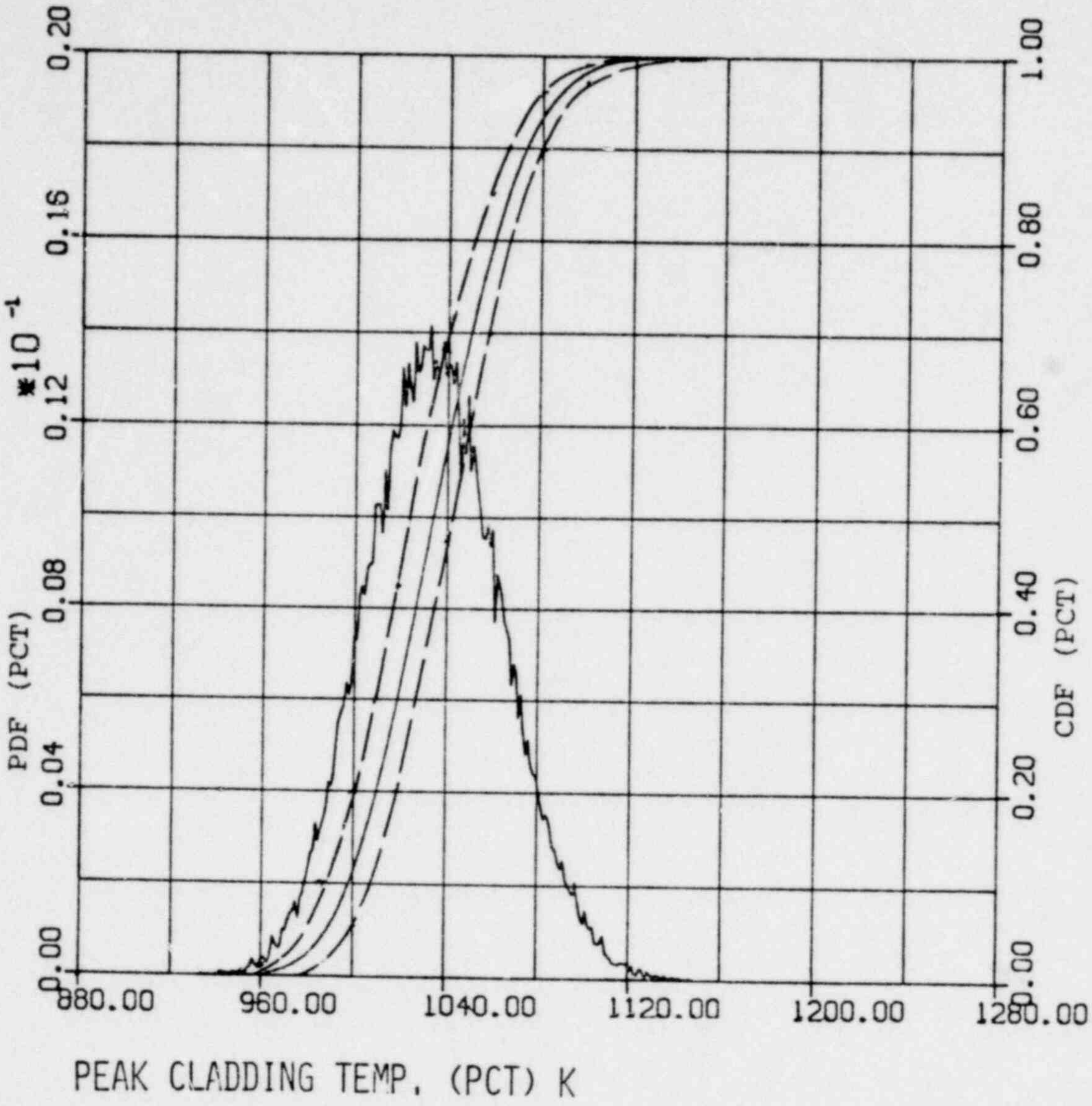

Fig. 2: pdf and cdf (with 95% confidence limits) for the peak cladding temperature of a fuel rod of high nominal rod power

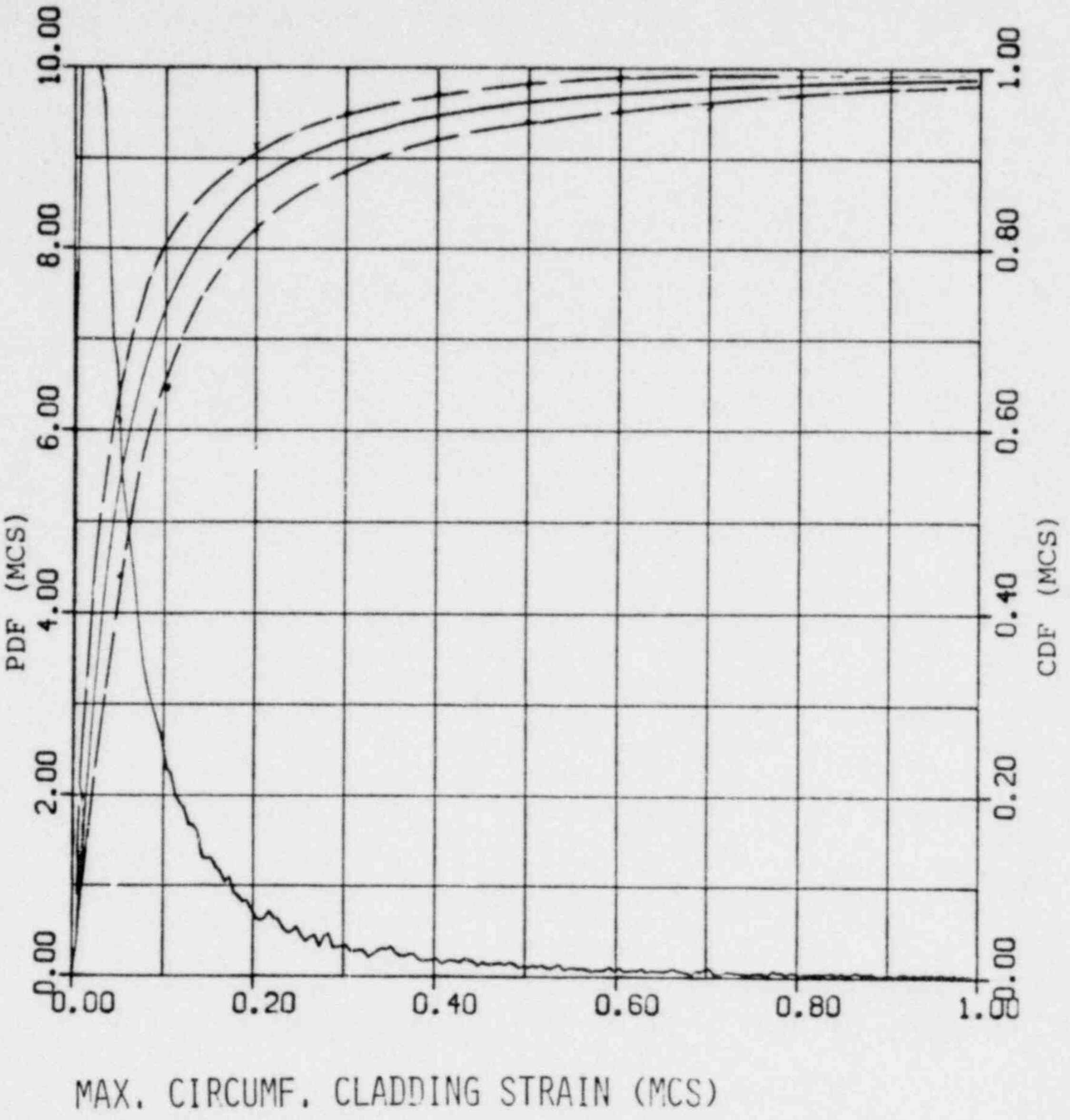

Fig. 3: pdf and cdf (with 95% confidence limits) for the maximum circumferential cladding strain of the same fuel rod referenced in Fig. 2.

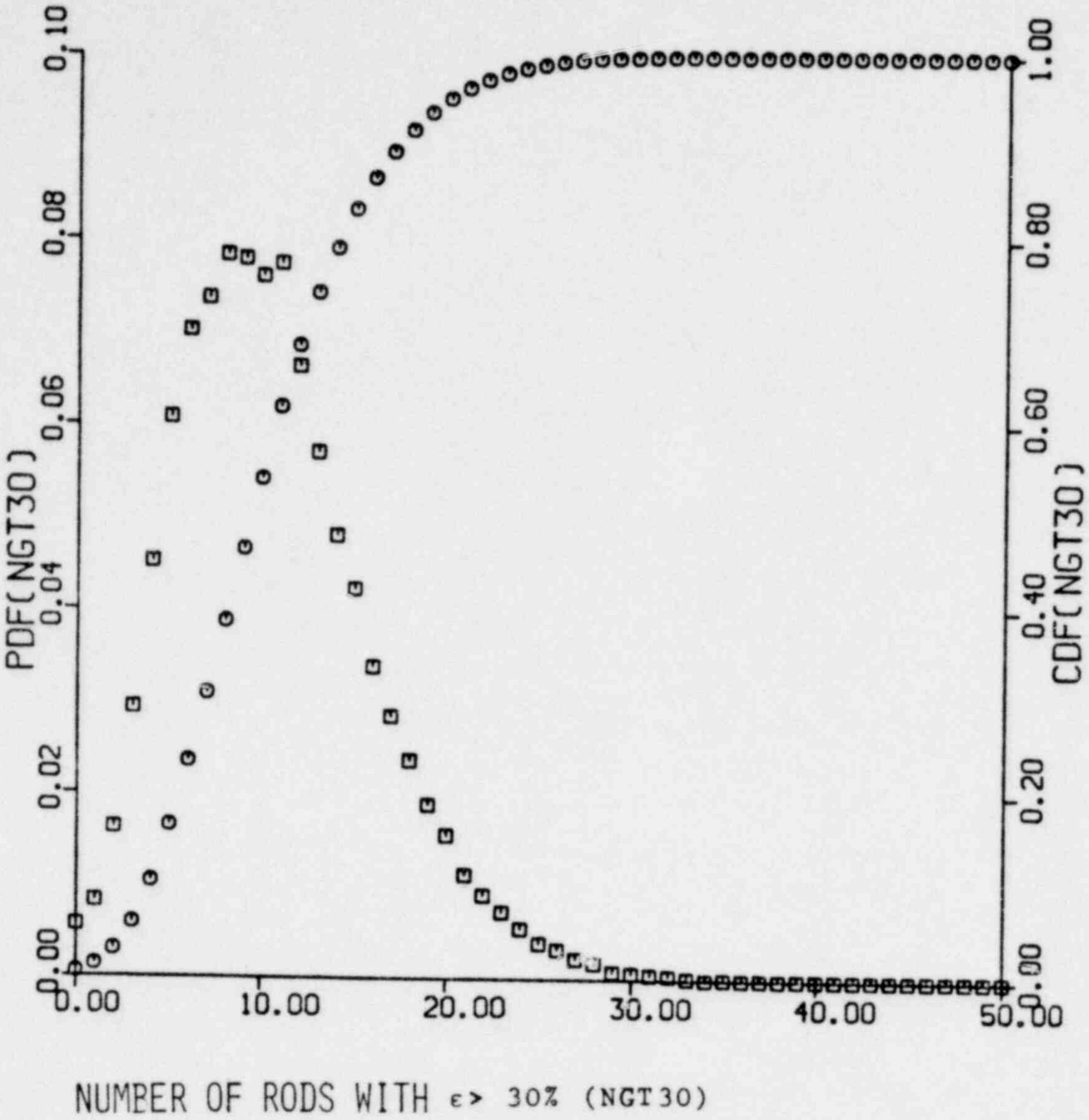

Fig. 4: pdf and cdf for the number of rods with cladding strains ε>30% within a PWR fuel rod bundle (236 fuel rods)

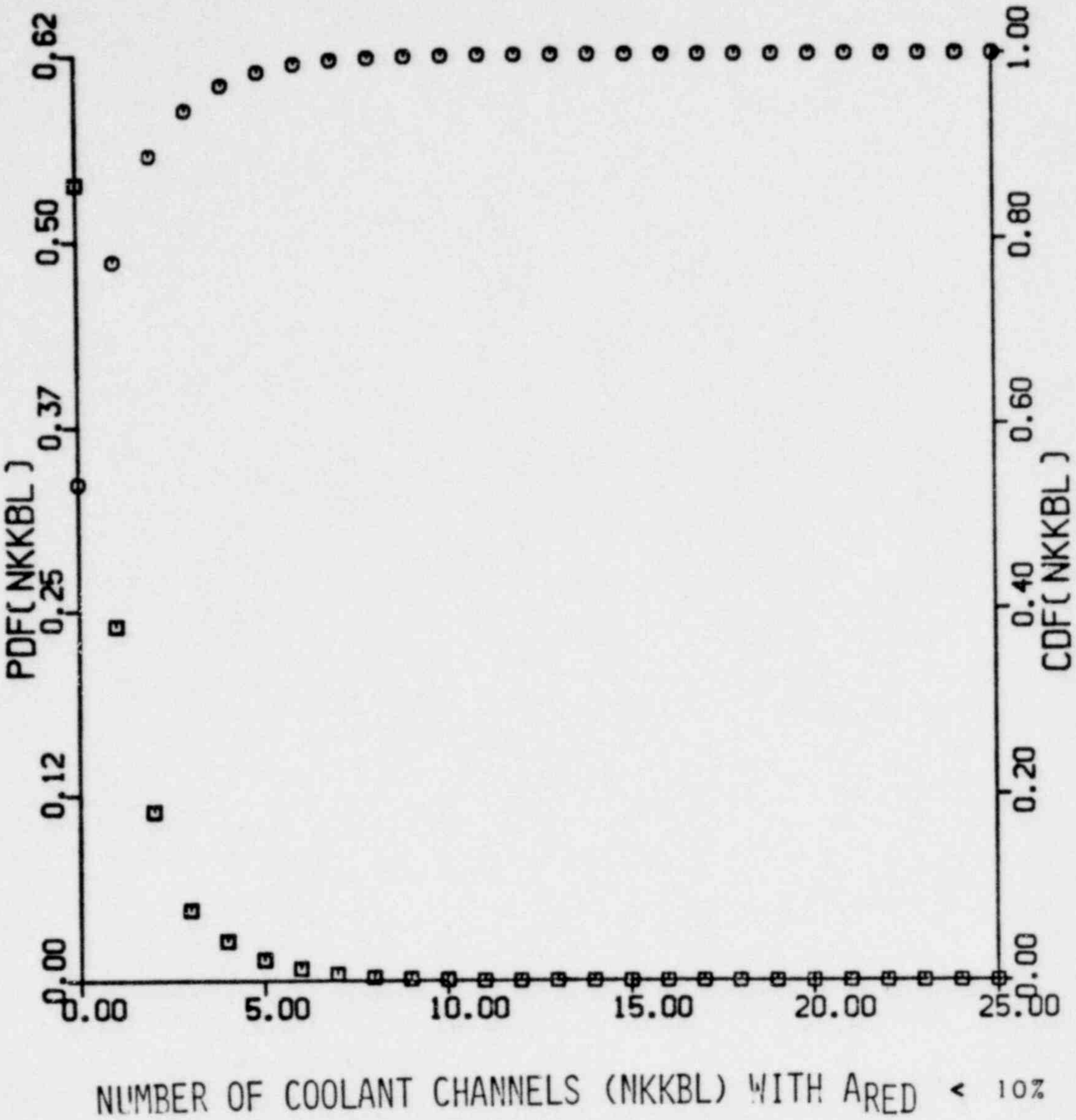

Fig. 5: pdf and cdf for the number of strongly reduced coolant channel areas (<10%) within the PWR fuel rod bundle

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