COMBUSTION ENGINEERING, INC.

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C-E METHODS FOR
LOSS OF FLOW ANALYSIS

Thermal Margin Engineering

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ABSTRACT

This report describes the assumptions, conservatisms and basic methods used for analyzing loss of reactor coolant forced flow events. The main body of the report describes a loss of flow analysis method for use with a computer code having transient core thermal hydraulic capabilities (referred to as the dynamic method). The appendix describes a similar loss of flow analysis method for use with a steady state core thermal hydraulic code (referred to as the static method). Sample analyses of the dynamic and static methods are presented using the CEDNBR and COSMO thermal hydraulic computer codes, respectively. These sample analyses are presented to illustrate the procedure used to analyze a loss of flow event. Current Combustion Engineering practice is to use the static method presented in the appendix with one exception: The TORC code and CE-1 CHF correlation are used instead of the COSMO code and W-3 correlation. It is concluded that the assumptions and methods presented herein constitute a conservative method of determining the consequences of a loss of flow event.

1.0 INTRODUCTION

This report describes a method for analyzing a loss of reactor coolant forced flow for use with a computer code having transient core thermal hydraulic capabilities (dynamic method). A similar method for use with a steady state core thermal hydraulic code (static method) is provided in the appendix. Sample calculations are provided to illustrate the analysis procedure. Current C-E practice is to use a variation of the static method presented in the appendix. The comments and conclusions presented in this report are applicable to any C-E nuclear steam supply system(NSSS).

The loss of reactor coolant forced flow (LOF) due to normal coastdown of one or more reactor coolant pumps is an anticipated operational occurrence (AOO). An AOO is by definition an event which is expected to occur one or more times during the life of the plant. The LOF due to simultaneous coastdown of all reactor coolant pumps is one of the most limiting AOOs because it produces a rapid approach to specified acceptable fuel design limits (SAFDL). Thus, this transient can impose limitations on allowable operating conditions. The transient is analyzed in order to determine the required limiting conditions for operation (LCO) of the reactor and to demonstrate the adequacy of the reactor protective system to mitigate the consequences of such an occurrence to the extent that the criteria for this AOO are met.

The LOF due to sudden stoppage of one reactor coolant pump is a postulated event for which protection is provided by the reactor protective system although some fuel damage may occur. The transient is analyzed in order to assure that pertinent limits on radiological releases from the plant are not exceeded.

During the course of execution, CESEC obtains steady state and transient solutions to the set of equations which mathematically describe the physical models of the subsystems mentioned above. Simultaneous numerical integration of a set of nonlinear, first-order differential equations with time-varying coefficients is carried out by means of a predictor corrector Runge-Kutta scheme. As the time variable evolves, edits of the principal system parameters are printed at prespecified intervals. An extensive library of the thermodynamic properties of uranium dioxide, water, and zircaloy is incorporated into this program. Through the use of CESEC, symmetric and asymmetric plant responses over a wide range of operating conditions can be determined.

3.1.4 CEDNBR

CEDNBR is a digital computer code developed by Combustion Engineering which can be used to determine the hot channel DNBR as a function of time.

The code solves the one-dimensional conservation of mass, energy, and momentum equations and the equation of state for the fluid. Tabular, time dependent functions of inlet fluid enthalpies, normalized average channel inlet mass velocity, normalized axial heat flux distributions, radial peaking factors, and eddy current mixing factors are required input for the code. Transient effects are included in the calculation of enthalpy rise and fluid properties (i.e., all transient terms are included). The W-3 CHF correlation is applied to fluid conditions in the hot channel at each of the 20 axial locations to determine the approach to departure from nucleate boiling(DNB).

CEDNBR is not currently used or planned to be used in any licensing submittals. It is utilized in this report for sample LOF calculations to demonstrate the method of analysis for use with a code with transient core thermal hydraulic capabilities (dynamic method).

3.1.5 COSMO

COSMO is a steady state thermal hydraulic computer program used to calculate core flow distribution, pressure drop, and W-3 Departure from Nucleate Boiling Ratio (DNBR) for both open and closed channel type cores. Unlike CEDNBR, transient effects in the coolant are not considered. A complete description of the program can be found in the topical report CENPD-161[5] which was submitted for review by the NRC July 1, 1975. COSMO is only utilized in the static method of analysis described in Appendix A.

3.1.6 TORC

TORC is a thermal hydraulic computer program used to simulate the fluid conditions within the reactor core and predict CE-1 DNBR (Ref. 4). A complete description of the program can be found in Reference 5.

3.2 Methods of Analysis

The method of analyzing a Loss of Flow described herein will be referred to as the dynamic method and consists of two major parts- (1) NSSS Response and Hot Channel DNBR calculation and (2) Radiological Consequences calculation.

The radiological consequences calculation is only explicitly performed when the computed minimum hot channel DNBR for the transient is less than 1.3, as in the seized shaft accident. This report will concern itself mainly with computation of NSSS response and minimum transient DNBR although the procedure involved in the radiological release calculation is described for completeness.

The dynamic method of calculating minimum DNBR for an LOF differs from the analytical method used by C-E for licensing submittals to date primarily in the method used to calculate the hot channel minimum DNBR. The dynamic method uses a code with transient core thermal hydraulic capabilities for this calculation. The method used for earlier licensing analyses utilizing the COSMO code and W-3 correlation is described in Appendix A and is referred to as the static method. Examples of the static method are presented using the COSMO code for the hot channel minimum DNBR calculation.

3.2.1 NSSS Response and Hot Channel DNBR

This section outlines the method of calculating the NSSS transient response and hot channel conditions during an LOF. Of particular interest is the minimum value of DNBR reached in the hot channel during the transient. The CEDNBR code is used to illustrate the procedure. A schematic of the calculational steps is given in Figure 3.2. The steps are as follows:

a. The transient coolant flow in each of the primary loops as a function of time is calculated by the COAST code. If measured plant data on flow coastdown is available for the plant under consideration, then either the measured data or a COAST calculated flow coastdown which underpredicts the measured coastdown will be used in the analysis.

b. The axial power distribution and associated CEA reactivity as a function of insertion is generated by the QUIX code. The scram reactivity as a function of insertion is used in normalized form by CESEC. A suitably conservative value is used for total CEA shutdown worth.

c. The response of average NSSS parameters (e.g., primary and secondary pressures, loop coolant temperatures, valve actions, etc) to the LOF is calculated by the CESEC code.

d. The normalized core average heat flux transient to be used to calculate the time of minimum DNBR in the hot channel is calculated by the CESEC code. This calculation is performed with the primary system pressure and the core inlet temperature held constant at their initial values.

e. The time at which the hot channel minimum DNBR is reached is calculated by a core transient thermal hydraulic code (e.g., CEDNBR) using the transient coolant flow through the core and the normalized heat flux calculated by steps a and d, respectively. This calculation uses the procedure of equating the time dependence of the normalized hot channel heat flux at any axial position to the time dependence of the normalized core average heat flux. This calculation is performed with the primary system pressure and the core inlet temperature held constant at their initial values.

7.0 CONCLUSIONS

It is concluded that the basic method in this report represents a conservative means of analyzing a loss of flow event utilizing a core thermal hydraulic computer code with transient capabilities. This conclusion applies to both anticipated operational occurrences and postulated accidents. The conclusion is supported by consideration of the conservative assumptions and procedures presented in Section 3.2 and is valid independent of the specific core thermal hydraulic computer code used. The sample results presented in Table 6.1 indicate that use of the dynamic method utilizing a core thermal hydraulic code with transient capabilities would provide an increase in reactor operational flexibility when compared to use of the static method of analysis.

8.0 REFERENCES

1. "COAST Code Description", CENPD-98, April, 1973.

2. CESSAR, Combustion Engineering Standard Safety Analysis Report, Docket No. STN-50-470.

3. "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", CENPD-107, C-E Proprietary Report, April, 1974.

4. "C-E Critical Heat Flux", CENPD-162-P-A, September, 1976.

5. "TORC, A Computer Code for Determining the Thermal Margin of a Reactor Core", CENPD-161, C-E Proprietary Report, July, 1975.

6. "Loss of Coolant Flow Analysis with One-Dimensional Space-Time Neutronics and Thermal Hydraulics", CENPD-111, C-E Proprietary Report, August, 1973.

7. NRC, Code of Federal Regulations (CFR), Title 10, Part 100, Reactor Site Criteria.

8. "PDQ-7, Reference Manual", WAPD-TM-678, January, 1968.

9. "Arkansas Nuclear One Unit 2 Final Safety Analysis Report", Docket No. 50-368.

APPENDIX A

STATIC METHOD OF ANALYZING AN LOF

A.1 INTRODUCTION

This appendix describes a method (referred to as the static method) for analyzing a loss of reactor coolant forced flow event utilizing a steady state core thermal hydraulic computer code. Sample calculations are provided to illustrate the analysis procedure. Current C-E practice is to utilize the static method presented herein with one exception: the TORC computer code and CE-1 CHF correlation are used instead of the COSMO code and W-3 correlation.

A.2 THE LOSS OF FLOW TRANSIENT

The causes of an LOF, in addition to the relevant safety criteria and RPS protection afforded, are described in Section 2.0 of the main body of this report.

A.3 ANALYSIS OF LOF TRANSIENTS

A.3.1 Computer Codes

A general description and reference are given in Section 3.1 of the main body of this report for the computer codes utilized by the static method to analyze the consequences of a loss of flow.

A.3.2 Methods of Analysis

The method of analyzing a loss of flow described herein will be referred to as the static method and consists, like the dynamic method, of two major parts-(1) DNBR and NSSS Response Computation and (2) Radiological Consequences Calculation.

The fuel damage calculation was performed using the information displayed in Figures A-3, 5.16, and 5.17. Figure A-3 is a plot of minimum transient hot channel DNBR vs radial peaking factor-generated by performing DNBR computations for various radial peaks. Figure 5.16 is a plot of DNBR vs probability of DNB-taken from the W-3 correlation. Figure 5.17 shows the number of fuel pins in the core having any given radial peak generated by various PDQ-7 computations[8]. The calculation of total integrated fuel damage percentage is illustrated in Table A-2.

A.6 CONCLUSIONS

It is concluded that the static method of analysis of an LOF presented in this Appendix provides an adequately conservative means of determining the consequences of LOF transients. This conclusion applies to both anticipated operational occurrences and postulated accidents and is valid independent of any reliance on the specific core thermal hydraulic computer code used. The conclusion is clearly supported by consideration of the conservative assumptions and procedures discussed in Section 3.2