OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN UNIT 1

CONTROL ELEMENT ASSEMBLY EJECTION ACCIDENT METHODOLOGY SUMMARY REPORT

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1.0 METHOD OF ANALYSIS

A complete description of the Westinghouse analysis methodology for the CEA ejection event is described in Reference 1. The methodology described within this report has been approved by the NRC for numerous applications on Westinghouse plants as well as for a Combustion Engineering plant loading Westinghouse fuel. Also found in Reference 1 are numerous sensitivity studies performed which provide the basis for the conservative choice of core physics characteristics used in this analysis. A brief discussion of the methodology found in Reference 1 follows.

The calculation of the CEA ejection event is performed in two stages. First, an average core channel calculation is done using TWINKLE; and then, a hot spot analysis is done using FACTRAN.

The average core calculation is performed using spatial neutron kinetics to determine the average power generation with time, including the various core reactivity feedback effects, i.e., Doppler and moderator reactivity. The nuclear power increase during this transient will lead to elevated fuel pellet and fuel cladding temperatures. The TWINKLE code is utilized, in conjunction with Fort Calhoun Unit 1 plant-specific physics data, to perform a one-dimensional (axial) average core neutron kinetic analysis allowing for a more realistic representation of the spatial effects of axial moderator feedback and CEA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods of calculating the CEA worth and hot channel peaking factor as discussed below.

The resulting average core nuclear power transient is input to FACTRAN along with the appropriate parameters such as fuel geometry, initial power, nominal average heat flux and core flow rate, initial and final hot spot total peaking factors, pellet power distribution, and gap heat transfer coefficients vs. time. Enthalpy and temperature transients in the hot spot are determined by multiplying the average core energy generation by the hot channel peaking factor and performing a fuel rod transient heat transfer calculation. During the transient, the steady-state heat flux hot channel factor is linearly increased to the transient value in 0.05 second, the assumed time for full ejection of the CEA. Prior to ejection, the power in this region will be depressed. However, the assumption is made that the hot spots before and after ejection are at the same axial location. This is conservative since the peak power after ejection will occur in or adjacent to the assembly with the ejected CEA.

In the hot spot analysis, the transient temperature distribution in a cross section of a metal clad uranium-dioxide fuel rod, and the heat flux at the surface of the rod, is calculated, using as input, the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature.

The FACTRAN computer code uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation after DNB. Prior to DNB, the code automatically selects between the forced convection (Dittus-Boelter) and local boiling (Jens-Lottes) correlations based on the clad temperatures calculated by each. The Bishop-Sandberg-Tong correlation is conservatively used, assuming zero bulk fluid quality. The DNBR is not calculated; instead, for the full power cases, the code is forced into DNB 0.05 seconds after the start of the transient while in the zero power cases, the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes.

Four cases are considered for this event to cover the spectrum of power levels and reactivity conditions which can occur throughout the fuel cycle. Full-power and zero-power cases are analyzed with reactivity coefficients consistent with end-of-life and beginning-of-life core physics conditions.

2.0 Computer Codes

2.1 TWINKLE

The TWINKLE code is a neutron kinetics code which solves the multidimensional, two-group transient diffusion equations using a finite-difference technique. The code contains a detailed six-region fuel-clad-coolant transient heat transfer model at each spatial point for calculating Doppler and moderator feedback effects. The method used to calculate feedback is similar to that used in Westinghouse nuclear design codes. TWINKLE handles up to 2000 spatial points in one-, two- or three-dimensional rectangular geometry and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, CEA motion and others to produce output of nuclear power as a function of time.

The TWINKLE code is used to predict the neutron kinetic behavior of a reactor core for transients, such as CEA ejection, which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 2.

2.2 FACTRAN

The FACTRAN computer code calculates the transient temperature distribution in a cross section of a metal clad, uranium dioxide fuel rod and the transient heat flux at the surface of the clad, using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature and density). The code uses a fuel model containing a sufficiently large number of radial space increments to model even fast transients. FACTRAN also uses material properties