ENCLOSURE

CORE PERFORMANCE BRANCH SAFETY EVALUATION OF VEPCO EVALUATION OF THE CONTROL ROD EJECTION TRANSIENT

VEP-NFE-2

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1.0 SUMMARY OF TOPICAL REPORT

This report describes the methods developed by the Virginia Electric and Power Company (Vepco) for the analysis of a postulated control rod ejection transient in the North Anna and/or Surry Nuclear Power Stations. A description of the rod ejection transient and a discussion of the acceptance criteria which must be met to assure the safe operation of the plant in the event of sucn a transient is also presented.

The RETRAN computer program is used for the analysis which is performed in two parts. First, a point kinetics analysis is used to calculate the average core nuclear power history. Next, a hot spot thermal-hydraulic calculation is used to determine the hot spot enthalpy and temperature transients from which the amount of fuel damage and radiological consequences are assessed. Detailed descriptions of the calculational model and the techniques employed in the analysis are presented.

The results of sensitivity studies used to quantify the effect of uncertainties in important core parameters and modeling assumptions on the model's predictions are shown. Also presented are comparisons of the results of the fuel vendor (Westinghouse) methodology with the Vepco methodology as well as comparisons of point kinetics results with three-dimensional space-time kinetics model results.

2.0 STAFF EVALUATION

We have reviewed the subject report, including the mathematical models and analytical procedures and methods. The RETRAN computer program is the principal calculational tool. A point kinetics analysis is used to calculate the average core nuclear power during a rod ejection transient. The hot spot (hottest fuel pin) enthalpy and temperature transients are determined from a hot spot thermal-hydraulic calculation. This is the usual procedure used by the nuclear industry to analyze the spatially dependent transient with a point kinetics model and has been found to be acceptable and usually conservative.

Although the RETRAN program is used to analyze the rod ejection transient, the report emphasizes the implementation of the RETRAN models since the detailed code description is available in a separate document. Therefore, we did not review the RETRAN program, per se, but rather the qualification of its use in determining the consequences of a rod ejection transient.

The staff position, as well as that of most of the reactor vendors and licensees, has been to limit the average fuel pellet enthalpy at the hot spot following a rod ejection transient to 280 calories per gram (cal/gm). This was based primarily on the results of the SPERT tests which showed that, in general, fuel failure consequences for UO_2 have been insignificant below 300 cal/gm for both irradiated and unirradiated fuel rods as far as rapid fragmentation and dispersal of fuel and cladding into the coolant are concerned. In this report, Vepco has chosen more stringent design limits. The limiting fuel failure criterion has been reduced to 225 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods. Since this is a conservative revision, the staff finds these criteria acceptable.

Vepco proposes a clad temperature limitation of 2700°F as the temperature above which clad embrittlement may be expected. Although this is several hundred degrees above the maximum clad temperature limitation proposed in the ECCS criteria, the staff feels this is adequate in view of the relatively short time at temperature and the highly localized effect of this transient. The staff has no limiting temperature criterion for rod ejection transients.

The neutronics model including the reactivity insertion, neutron kinetics parameters, fuel and moderator temperature feedback, and reactor trip assumptions described in the report are in conformance with the recommendations of USNRC Regulatory Guide 1.77 and are acceptable. Since RETRAN uses a point kinetics neutronics model, the effect of locally peaked core flux shapes due to the rod ejection is not included in the Doppler reactivity calculation. Therefore, a power weighting factor (PWF) is used to modify the Doppler reactivity versus core average fuel temperature data in order to better approximate the local reactivity effects. The TWINKLE computer program is used by Vepco to justify the use of a PWF. Since TWINKLE is a three-dimensional spatial kinetics code, it accounts for the highly localized reactivity effects of a control rod ejection accident more realistically than a point kinetics calculation and is, therefore, an acceptable method for verifying the PWF. For cases initiated from both hot zero power and hot full power initial conditions, the RETRAN point kinetics predictions are shown to be conservative compared to the TWINKLE three-dimensional predictions and, therefore, the described RETRAN model is acceptable.

The radiological effects of a rod ejection transient have been addressed generically in the Surry 1 and 2 and the North Anna 1 and 2 Updated Final Safety Analysis Reports as well as in the Westinghouse "Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods" (WCAP-7588). In this latter report, a departure from nucleate boiling (DNB) analysis was performed using the transient THINC-III code for a worst-case three-dimensional rod ejection transient. The results indicated that less than 10% of the fuel rods enter DNB. Since one of the NRC acceptance criteria for this transient requires the assumption that all of the fuel rods which experience DNB release their entire gap inventory of fission products to the coolant, the generic result indicates that less than 10% of the core will release fission products. The source term for the radiological release calculations for these and other Westinghouse designed plants is, therefore, based on 10% of the fuel experiencing clad failure and releasing their entire gap activity. This is in conformance with Regulatory Guide 1.77 and, hence, acceptable.

Since there is no fuel dispersal into the coolant, the pressure surge during the transient may be calculated based on conventional heat transfer methods. Generic pressure surge calculations for the most severe excess addition of energy to the coolant indicate that any system overpressurization due to a rod ejection transient will meet the NRC criterion of being less than that pressure which would cause stresses to exceed the Service Limit C as defined in Section III of the ASME Boiler and Pressure Vessel Code.

Vepco has analyzed the rod ejection transient using Westinghouse codes and techniques and compared their results to the results obtained by Westinghouse. We have reviewed these comparisons and concur that Vepco can adequately replicate Westinghouse results. We have also reviewed comparisons between results obtained with the NRC approved Westinghouse methodology with those obtained with the Vepco methodology presented in the topical report using RETRAN. We concur that these comparisons demonstrate the acceptability of the Vepco methodology. A comparison is also presented between the Vepco calculated results using both Westinghouse and Vepco methodologies and results obtained by Vepco using the Westinghouse three-dimensional space-time kinetics code, TWINKLE. The staff finds that this comparison adequately demonstrates the conservatism of the Vepco rod ejection calculational method.

3.0 EVALUATION PROCEDURE

The staff has reviewed the report within the guidelines provided by Sections 4.3 and 15.4.8 of the Standard Review Plan (NUREG-75/087) and by Regulatory Guide 1.77. Part of our review was based on our familiarity with and comparison of similar analyses for control rod ejection transients provided in topical reports by other PWR vendors. The staff also had the benefit of a meeting with VEPCO concerning the report.

4.0 REGULATORY POSITION

The subject report (VEP-NFE-2) provides an acceptable method for analyzing the control rod ejection event for Vepco's Surry and North Anna Nuclear Power Stations. This methodology may be referenced in future Vepco license applications.

SALP INPUT FOR VEP-NFE-2 TOPICAL REPORT

CORE PERFORMANCE BRANCH

The following two functional areas are applicable to our review of the topical report:

1. Management Involvement.

   The presentation to the NRC and the information contained in the topical report shows well stated and thorough information relevant to the subject matter.

   Rating: Category 1

2. Approach to Resolution of Technical Issues.

   The documentation demonstrates a clear understanding of the required assumptions and calculational methods and exhibits conservatism in those assumptions and methods.

   Rating: Category 1