SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MINOR REVISION TO TOPICAL REPORT DPC-NE-1004A

DUKE POWER COMPANY

DOCKET NOS. 50-413, 50-414, 50-369 and 50-370

1.0 INTRODUCTION

By letter dated December 12, 1995, as supplemented on April 9, 1996, (Reference 1), Duke Power Company, (DPC or licensee), requested review of a revision to topical report DPC-NE-1004A, (Reference 2). Specifically, DPC requested review of an effort to improve its reload design methods. The CASMO-3/SIMULATE-3P power distribution uncertainty factors have been re-evaluated by DPC using measured data from recent Catawba and McGuire fuel cycles. The revision includes increasing the axial nodes in SIMULATE-3P from 12 axial nodes to 24 axial nodes. This change in the number of nodes structure will allow explicit modeling of axial blanket fuel segments.

Benchmarking calculations were performed and presented in tabular form in the DPC submittal.

2.0 EVALUATION

The results (Observed Nuclear Reliability Factors, ONRFs) of the calculations for the assembly radial, axial, and total peaking factors (using 24 nodes), were compared to those results obtained in the NRC approved topical, DPC-NE-1004A. The comparison showed that the axial and total peaking uncertainty factors decreased for the new (24 nodes) benchmarking and the assembly radial power peaking increased slightly from 1.017 to 1.020. The statistical analysis used in this analysis is the same as that used in the approved topical, DPC-NE-1004A. The benchmarking performed by the licensee included recent cycle data such as, longer cycle length, higher fuel enrichment, and consequences of additional burnable poisons. The cores used in the benchmarking were McGuire 1, McGuire 2, cycle 9, Catawba 1 cycle 7, and Catawba 2, cycles 6 and 7.

2.1 A Change in the Number of Nodes: from 12 Nodes to 24 Nodes.

The increase from 12 nodes to 24 nodes will remove the coding limitation (number of axial regions which can be modeled in one node) in SIMULATE-3P. In order to take advantage of axial blanket fuel with burnable poisons rod, a 6 inch axial node was chosen, because nodal boundaries match exactly with the transition from the blanketed fuel region to the non-blanketed fuel region. Boundary matching enhances calculational accuracies and simplifies cross-section assignments. More importantly, the increase in the number of nodes will provide a more accurate prediction of measured power distributions of current generation vendor fuel designs which are typically not axially homogeneous.

The original ONRFs for the Westinghouse plants were developed in the approved topical DPC-NE-1004 based on the analysis of the McGuire 2 cycles 4 and 5, Catawba 1, cycle 3 and Catawba 2 cycle 2 core designs. The 24 nodes ONRFs were developed based on current core designs, reflecting the more aggressive reload design strategies reflected in current core designs. Included in current core design databases is such information as higher enrichment, higher burnup, longer cycle lengths, and axial loading.

The analysis conducted by the licensee indicated that the F_q and F_z ONRF's decreased by 2.0 and 2.2%, respectively, relative to ONRF's stated values in DPC-NE-1004A. The minor statistical increase in the radial axial peaking factor, $F_{\Delta H}$, is due mainly to the increase in the number of nodes which contributes to a more radially heterogenous core design.

The decrease in the F_q and F_z uncertainties is due to the increase in the accuracy of presentation of the axial spectral dependency, as a result of the reduction of axial node size.

These reductions (decreases in the uncertainties) result in a reduction in the bias term included in the ONRF.

The licensee analyzed the impact of the increase in the $F_{\Delta H}$ uncertainty and found that the increase in the $F_{\Delta H}$ was mainly due to the increase in the complexity of the reactor cores and very little of it was due to the increase in the number of nodes. Consequently, the licensee considered the impact of this increase on the FSAR Chapter 15 accidents. The Chapter 15 calculations that were effected are pin pressure, creep collapse, and DNB. Peak fuel enthalpy, linear heat rate to melt limits, and primary and secondary peak pressures are not affected by the increase in radial uncertainty factors, because the licensee showed that in each case, either significant margin exists, (and thus the increase in the radial uncertainty factor is not limiting), or the actual pin peak pressure and creep calculations assume a bounding uncertainty value higher than that for the 24 axial node value.

The licensee conducted Chapter 15 DNB analysis to ensure that fuel integrity is maintained and that the minimum DNBR remains above the 95/95 DNBR limit. They conducted two kinds of DNBR analyses: 1) A thermal analysis which is based on the Statistical Core Design (SCD) methodology described in (Reference 3), and 2) a thermal analysis which is not based on an SCD method.

For the first method, the pertinent Chapter 15 accidents were not affected because bounding radial and axial uncertainty factors which are greater than those of the 24 node analysis are assumed in the final analyses. For the second method, Chapter 15 analyses including Startup of Inactive RC pump, Steam Line Break, Locked Rotor, and Rod Ejection, were analyzed separately.

For the startup of the inactive coolant pump, and of the steam line break, the $F_{\Delta H}$ assumed bounds both the 12 and 24 axial level uncertainties. For the rod ejection accident, analysis showed that the present $F_{\Delta H}$ uncertainty did not bound the $F_{\Delta H}$ uncertainty calculated for the 24 axial level model. However, analyses conducted by the licensee showed that, since DNB is a function of both the radial and axial power distributions, the decrease in the 24 axial level uncertainty factor more than offsets the slight increase in the radial uncertainty, resulting in a net increase in the DNB margin. Consequently, it can be concluded that analyses regarding Locked Rotor and Steam Line Break are conservative, and that the margin of safety as defined in the Technical

Specifications are maintained. The staff agrees with these conclusions.

3.0 CONCLUSION

The NRC staff has reviewed the revision, dated December 12, 1995 and April 9, 1996, to Topical Report DPC-NE-1004A, submitted by the Licensee for the operation of the Catawba and McGuire Nuclear Stations. Based on this review, the staff concludes that the requested minor change to the above mentioned topical is acceptable.

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4.0 REFERENCES

1. Duke Power Company Letter, M. S. Tuckman, dated December 12, 1995, to U.S. Nuclear Regulatory Commission, Document Control Desk, Washington DC 20555.

2. Duke Power Company Letter, M. S. Tuckman, Dated April 9, 1996, Responses to Request for Additional Information.

3. DPC-NE-2005P-A, February 1995, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology."