DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT DPC-NE-1005

NUCLEAR DESIGN METHODOLOGY USING CASMO-4/SIMULATE-3 MIXED OXIDE (MOX)

CATAWBA AND MCGUIRE NUCLEAR STATIONS, UNITS 1 AND 2

1.0 INTRODUCTION

The Duke Power Company (Duke or licensee) submitted by letters dated August 3 (Proprietary), and August 6, 2001 (Non-proprietary), the Topical Report DPC-NE-1005P, Revision 0, "Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX" for review by the Nuclear Regulatory Commission (NRC) staff. Duke is the licensee for the Catawba Nuclear Station, Units 1 and 2 (Catawba), and the McGuire Nuclear Station, Units 1 and 2 (McGuire). Duke submitted additional letters dated September 12, and November 12, 2002, June 26, August 14 and December 2, 2003 (References 2, 3, 4, 5 and 6).

The Topical Report addresses the use of the Studsvik Core Management System (Studsvik/CMS) code package to support the reload design analyses for Catawba and McGuire. The Studsvik/CMS code package primarily consists of the CASMO-4 and SIMULATE-3 MOX computer codes. The Topical Report demonstrates the validity and accuracy of the CMS package at Catawba and McGuire for core reload design, core follow, and the calculation of key core parameters for reload safety analysis. The NRC staff's review of the report considered the report's applicability for the use of low enriched uranium (LEU) fuel at Catawba and McGuire and the use of up to four Lead Test Assemblies (LTA) in one of the Catawba units. The NRC staff's review findings are based, in part, on licensee commitments included by Duke in Reference 4 as follows:

1. For a lead assembly program containing four MOX fuel assemblies, Duke will place at least two of the MOX fuel lead assemblies in core locations that are measured directly by the movable incore detector system for the first and second cycles of lead assembly irradiation.

2. Duke will perform the physics test program defined in Table 1 [of Reference 4] for all MOX fuel lead assembly cores and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined [in Reference 4] is reached. Core power levels at which low and intermediate power escalation power distribution maps are taken will be consistent from cycle to cycle for each unit (within ± 3% rated thermal power). Core power level at which power distribution maps are taken may vary among units and between McGuire and Catawba.

3. Duke will prepare a startup report for each operating cycle with MOX fuel lead assemblies and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined in Reference 4 is reached. Each

startup report will contain comparisons of predicted to measured data from the zero power physics tests and the power distribution maps taken during power escalation. The reports will include discussions of any parameter that did not meet acceptance criteria. Duke will provide each report to the NRC within 60 days of measurement of the final power distribution map.

4. Duke will prepare an operating report for each operating cycle with MOX fuel lead assemblies and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined in Reference 4 is reached. Each operating report will contain comparisons of predicted to measured monthly power distribution maps and monthly boron concentration letdown values. Duke will provide each cycle operating report to the NRC within 60 days of the end of the fuel cycle.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 34, “Contents of Applications; Technical Information,” requires that Safety Analysis Reports be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload process, licensees perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. Licensees confirm that the analyses remain bounding by ensuring that the inputs to the safety analyses are conservative with respect to the current design cycle. They check these inputs by using core design codes and methodologies.

The objective of the nuclear design review for the fuel assemblies, control systems, and reactor core is to aid in confirming that fuel design limits will not be exceeded during normal operation or anticipated operational transients. The NRC staff acceptance criteria are based on Chapter 4.3, “Nuclear Design” of the Standard Review Plan.

3.0 TECHNICAL EVALUATION

Currently, the Catawba and McGuire Nuclear Stations use the CASMO-3/Simulate-3 analytical computer codes and various methodologies. In its submittal, Duke requests replacing its current codes with the newer Studsvik/CMS package. The CASMO-4, CMS-LINK, SIMULATE-3 MOX, and SIMULATE-3K MOX computer codes comprise the Studsvik/CMS package.

The CASMO-4 computer code is the Studsvik Scandpower, Inc. lattice code. The CASMO-4 computer code, a multi-group two-dimensional transport theory code for depletion and branch calculations for a single assembly, is used to generate the lattice physics parameters. These parameters include the cross sections, nuclide concentrations, pin power distributions and other nuclear data used as input to the SIMULATE-3 MOX program for core performance analyses.

New features of CASMO-4 over CASMO-3 are the incorporation of the microscopic depletion of burnable absorbers into the main calculations, and the introduction of a heterogeneous model for the two-dimensional calculation. Also new in CASMO-4 is the use of the characteristics method for solving the transport equation. When MOX fuel is detected in the input, the code automatically uses a more detailed internal calculation to accommodate the larger variation of plutonium cross sections and resonances. Studsvik also supplies the SIMULATE-3 MOX code. This code is a two-group, 3-dimensional nodal program based on the NRC staff approved QPANDA neutronics model that employs fourth-order polynomial representations of the intranodal flux distributions in both the fast and thermal neutron groups. The code is based on modified coarse mesh (nodal) diffusion theory calculational technique, with coupled thermal hydraulic and Doppler feedback. The program explicitly models the baffle/reflector region, eliminating the need to normalize to higher-order fine mesh calculations. It also includes the following modeling capabilities: solution of the two group neutron diffusion equation, fuel assembly homogenization, explicit reflector cross-section model, cross-section depletion and pin power reconstruction. The SIMULATE-3 MOX code uses a more refined solution technique to account for steeper flux gradients that exist between the MOX and LEU fuel interfaces.

In order to insure flux continuity at nodal interfaces and perform an accurate determination of pin-wise power distributions, SIMULATE-3 MOX uses assembly discontinuity factors that are pre-calculated by CASMO-4. These factors are related to the ratio of the nodal surface flux in the actual heterogenous geometry to the cell averaged flux in an equivalent homogeneous model, and are determined for each energy group as a function of exposure, moderator density and control rod state.

The two group model solves the neutron diffusion equation in three dimensions, and the assembly homogenization employs the flux discontinuity correction factors from CASMO-4 to combine the global (nodal) flux shape and the assembly heterogeneous flux distribution. The flux discontinuity concept is also applied to the baffle/reflector region in both radial and axial directions to eliminate the need for normalization, or other adjustment at the core/reflector interface.

The SIMULATE-3 MOX fuel depletion model uses tabular and functionalized macroscopic or microscopic, or both cross sections to account for fuel exposure without tracking the individual nuclide concentrations. Depletion history effects are calculated by CASMO-4 and then processed by the CMS-LINK code for generation of the cross section library used by SIMULATE-3 MOX.

SIMULATE-3 MOX can be used to calculate the three dimensional pin-by-pin power distribution in a manner that accounts for individual pin burnup and spectral effects. SIMULATE-3 MOX also calculates control rod worth and moderator, Doppler and xenon feedback effects.

3.1 Model Benchmarking

The licensee’s submittal, dated August 3, 2001, compares the CASMO-4/SIMULATE-3 MOX predictions of key physics parameters against plant data and critical experiments. For CASMO-4, this benchmarking encompassed criticality and pin power predictions for LEU and MOX fuel. As part of the development of the Catawba and McGuire models, the licensee compared CASMO-4/SIMULATE-3 MOX calculation predictions to plant and/or experimental data for reactivity worth for soluble boron, burnable poison rods, silver-indium-cadmium control rods, Isothermal temperature coefficient, and core power distribution. The licensee provided documentation that contained the results of benchmarking CASMO-4 results to Monte Carlo code calculations and critical experiments for LEU and MOX fuel assembly designs (References 7 and 8).

The licensee performed comparisons between CASMO-4 MOX predictions and data from three MOX critical experiments: Saxton, EPICURE, and ERASME/L. The results of these comparisons were used in the development of the fuel pin power uncertainties that are part of the overall nuclear uncertainty factors. The Saxton critical experiment used plutonium that had an isotopic content that is close to current weapons grade plutonium fuel. EPICURE used fuel pins that are similar to current 17x17 pressurized-water reactors fuel pins and emulated the hot condition fuel to moderator ratio. ERASME/L used a fissile plutonium concentration of 8.23 percent that bounds the fissile plutonium content expected in the Duke reactors. SIMULATE-3 MOX could not model the experiments because of their small configurations; therefore, theoretical problems were developed to test the ability of SIMULATE-3 MOX to replicate the CASMO-4 calculations. This provides greater assurance that the CASMO-4/SIMULATE-3 MOX suite of codes will predict the core parameters for a core containing four MOX LTAs with acceptable accuracy.

The comparison of CASMO-4/SIMULATE-3 MOX predictions to measured data incorporates bias and uncertainty for both the predictions and the measured data. The licensee then used statistical methods to account for these uncertainties. For MOX fuel, these methods accounted for the uncertainty from the CASMO comparisons with data and the uncertainty from the CASMO to SIMULATE comparisons for the theoretical problems. Duke also used the CASMO/SIMULATE predictions in combination with the normalized flux map reaction rate comparisons to determine appropriate peaking factor uncertainty factors.

Duke intends to use the CASMO-4/SIMULATE-3 MOX programs in licensing applications, including calculations for core reload design, core follow, and calculation of key core parameters for reload safety analyses of Catawba and McGuire Nuclear Stations, Units 1 and 2. The licensee used data from the Catawba, Unit 1, operating cycles 11 through 13, Catawba, Unit 2, operating cycles 9 through 11, McGuire, Units 1 and 2, operating cycles 12 through 14 to benchmark the CASMO-4/SIMULATE-3 MOX models for LEU fuel. Duke also used data from the St. Laurent reactor in France, cycles 5 through 10, to benchmark the CASMO-4/SIMULATE-3 MOX models for MOX fuel. These cycles cover core design changes over 17 cycles of operation. Comparison of the St. Laurent parameters to the Catawba and McGuire reactor parameters were provided and demonstrated that the fuel and core parameters important to predicting the core physics response were similar. Loading pattern variations include out-in and low-leakage designs. For model benchmarking, the licensee used critical boron concentration measurements, startup physics testing data, estimated critical position information, and flux maps. The good agreement between the measured and the calculated values presented in the August 3, 2001, submittal, is used to validate the Duke application of these computer programs for analysis of the Catawba and McGuire Nuclear Stations Units 1 and 2 for LEU and a MOX LTA (maximum 4 LTA assemblies in one of the Catawba Nuclear Station) cores.

For the parameters compared, the licensee calculated a sample mean and standard deviation of the observed differences. They also determined bias to describe the statistical difference between predicted and reference values.

The St. Laurent reactor uses reactor grade MOX fuel and though similar in composition to the weapons grade MOX fuel, the isotopic composition differs slightly. The Saxton critical experiment uses a plutonium isotopic composition that is very close to the weapons grade MOX (90 percent fissile plutonium composition.) Both benchmarks demonstrate that the

CASMO-4/SIMULATE-3 MOX code can provide close predictions and provides confidence that the code will provide a close prediction of the MOX LTAs. To support future batch implementation, Duke provided a commitment in Reference 3 that at least two of the MOX LTAs will be placed in instrumented core locations so that the results from the startup physics tests can be compared to the CASMO/SIMULATE predictions to demonstrate the applicability of the codes to analyze LEU/MOX fueled cores. The results of these benchmarks will be submitted to the NRC for review and approval.

The licensee demonstrated that the CASMO-4/SIMULATE-3 MOX models, in conjunction with the indicated reliability factors adequately represent the operating characteristics of the Catawba and McGuire Nuclear Stations. Additionally, Duke did not change key aspects of its core design and analysis methodology, and maintains code and quality assurance practices that provide assurance that future changes to the core, fuel, and burnable poison design will be modeled with accuracy and conservatism. Since the Studsvik/CMS package adequately represents the operating characteristics, the NRC staff finds the use of the Studsvik/CMS package to be acceptable for Catawba for LEU fuel and up to four MOX LTAs and for McGuire with LEU fuel.

3.2 Statistics

The NRC staff reviewed Duke's application for statistical content. The statistical issues revolved around the 95/95 tolerance limit calculations for each parameter of interest. The calculations give 95 percent assurance that at least 95 percent of the population will not exceed the tolerance limit.

The procedure used in the tolerance limits depended on whether the data could be assumed to be distributed normally. The licensee used several established techniques for testing normality and assumed normality only if the majority of the techniques validated that assumption. This approach is acceptable to the NRC staff.

When the normal distribution was applicable, the licensee used the traditional one-sided tolerance calculations. Otherwise, the licensee used a nonparametric method to determine a conservatively large uncertainty (References 9, 10, 11 and 12). Both the parametric and the nonparametric approaches in their proper context are acceptable to the NRC staff.

3.3 Dynamic Rod Worth Measurement

Dynamic rod worth measurement (DRWM) provides a methodology for the licensee to measure the reactivity worth of the individual control rod banks without changing the boron concentration. The DRWM methodology takes the neutron flux signal from the excore detectors and conditions the excore detector signal through the use of analytical factors to convert the signal into the corresponding rod worth. The Safety Evaluation that approved the Westinghouse DRWM methodology required that anyone applying to use the methodology with their own codes perform calculations comparing their code results to the Westinghouse generated results and that the results must agree within 2 percent or 25 pcm for individual banks, and 2 percent for total bank worth. The acceptance criteria was developed to demonstrate that other parties that used the methodology were applying the codes and methodology correctly. The final test of using the methodology correctly is developing analytical factors that are consistent with the corresponding Westinghouse computations. This consistency is demonstrated by the measured rod worth comparisons.

Duke used the CASMO-4/SIMULATE-3 MOX codes to generate comparisons to the Westinghouse generated results that used the ALPHA/PHOENIX/ANC codes per the DRWM topical requirements. Duke's analysis showed that 3 percent of the computational results did not meet the criteria. All of the comparisons that did not meet the criteria were for predictions of the rod worth. The comparisons between the measured rod worth, CASMO-4/SIMULATE-3 MOX and the Westinghouse results demonstrated that the analytical factors developed using the CASMO-4/SIMULATE-3 MOX code very closely mirror the Westinghouse results. All of the measured rod worth comparisons met the acceptance criteria.

When the underlying causes of the computational results that did not meet the criteria were investigated, it was noted that the predicted and measured rod bank worth deviations were consistent with the differences in the predicted radial Hot Zero Power distribution between Westinghouse and Duke. Duke under-predicts the power of the assemblies on the core periphery, which results in a calculated lower rod worth for the associated rod banks (banks SA, CD, SD, and SC), and over-predicts the power of the assemblies in the center of the core, which results in a calculated higher rod worth for the associated rod banks (banks CC, CA, and SB). In all cases where the predicted rod worth computational results did not meet the criteria, Duke predicted a lower bank rod worth that was consistent with the radial power distribution difference between Westinghouse and Duke. Likewise, the impact of the radial distribution caused Duke to consistently calculate a lower total bank worth relative to the Westinghouse calculation since a greater number of rod banks are on the periphery.

The parameter of greatest interest for correct application of DRWM is the calculation of the analytical factor. Correct determination of the analytical factor is shown by close agreement in the measured rod worth comparisons. All of the measured rod worth comparisons met the acceptance criteria. Since all of the measured rod worth comparisons met the acceptance criteria and the deviations in the predicted rod worth comparisons were consistent with the radial power distribution predictions, the NRC staff finds the use of the CASMO-4/SIMULATE-3 MOX methodology acceptable for use with the DRWM methodology.

4.0 CONCLUSION

Duke submitted the Topical Report (Reference 1) and supplementary information in References 2, 3, 4, 5, and 6 for review by the NRC staff. The licensee performed extensive benchmarking using the CASMO-4/SIMULATE-3 MOX methodology. The licensee's effort consisted of conducting detailed comparisons of calculated key physics parameters with measurements obtained from several operating cycles of the Catawba and McGuire Nuclear Stations, Units 1 and 2, the St. Laurent reactor in France, and several MOX critical experiments. These results were then used to determine the set of 95/95 (probability/confidence) tolerance limits for application to the calculation of the stated physics parameters.

Based on the review of the analyses and results presented in References 1, 2, 3, and 4, the NRC staff has concluded that the CASMO-4/SIMULATE-3 MOX methodology, as validated by Duke, can be applied to the Catawba and McGuire steady-state physics calculations for reload applications as described in the above technical evaluation. The NRC staff's approval is limited to the range of fuel configurations and core design parameters as stated and referenced by the

August 3, 2001, submittal. Introduction of significantly different fuel designs will require further validation of the above stated physics methods for application to Catawba and McGuire by the licensee and will require review by the NRC staff. Additionally, the results of the LTA in-core performance and predictive capabilities of CASMO-4/SIMULATE-3 MOX for weapons grade MOX will need to be demonstrated and submitted to the NRC for review. This approval is subject to the conditions listed above in Section 1.0 that have been provided by Duke in Reference 4.

5.0 REFERENCES

1. Letter from K. S. Canady, Duke Power to the USNRC, “Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,” August 3, 2001 (Proprietary). A non-proprietary version was submitted by letter dated August 6, 2001.

2. Letter from K. S. Canady, Duke Power to the USNRC, “Response to Request for Additional Information - Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,” September 12, 2002.

3. Letter from K. S. Canady, Duke Power to the USNRC, “Response to Request for Additional Information - Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,” November 12, 2002.

4. Letter from M. S. Tuckman, Duke Power to the USNRC, “Physics Testing Program in Support of Topical Report DPC-NE-1005P, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,” June 26, 2003.

5. Letter from K. S. Canady, Duke Power to the USNRC, “Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,” August 14, 2003.

6. Letter from K. S. Canady, Duke Power to the USNRC, “Additional Information Related to Duke Topical Report DPC-NE-1005P, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,” December 2, 2003.

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