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MEMORANDUM FOR: Docket File

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SUBJECT: CATAWBA AND MCGUIRE NUCLEAR STATIONS -- FACSIMILE TRANSMISSION, DRAFT SAFETY EVALUATION ON TOPICAL REPORT DPC-NE-2009 (TAC MA 2359, MA2361, MA2411 AND MA2412)

The attached draft safety evaluation on Topical Report DPC-NE-2009, "Duke Power Company Westinghouse Fuel Transition Report," was transmitted by fax today to Mr. Steve Warren of Duke Energy Corporation to prepare him and others for a conference call. This memorandum and the attachment do not convey a formal request for information or represent an NRC staff position.

Docket Numbers 50-413 and 50-414
50-369 and 50-370

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. TO FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

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1.0 INTRODUCTION AND BACKGROUND

By letter dated July 22, 1998 (Ref. 1), and supplemented by a letter of October 22, 1998 (Ref 2), Duke Energy Corporation* (DEC, the licensee), the licensee for operation of McGuire and Catawba Nuclear Stations, proposed changes to the Technical Specifications (TS) of these plants in anticipation of a reactor core reload design using Westinghouse fuel. Accompanying the July 22, 1998, letter is a topical report DPC-NE-2009, "Duke Power Company* Westinghouse Fuel Transition Report," (Ref. 3) for NRC review and approval. When approved, this topical report will be listed in Section 5.6.5 of the Catawba and McGuire TSs as an approved methodology for the determination of the core operating limits.

The reactors of McGuire and Catawba Nuclear Stations are currently using Framatome Cogema Fuels (FCF) Mark-BW fuel assemblies (Ref. 4). The proposed amendment to the TSs would permit transition to the 17x17 Westinghouse Robust Fuel Assembly (RFA) design.

The RFA design is based on the VANTAGE+ fuel assembly design, which has been approved by NRC as described in WCAP-12610-P-A (Ref. 5). The RFA design to be used at McGuire and Catawba, as described in Section 2.0 of DPC-NE-2009, will incorporate the following features in addition to the VANTAGE+ design features:

- increased guide thimble and instrumentation tube outside diameter
- modified low pressure drop structural mid-grids
- modified intermediate flow mixing grids
- pre-oxide coating on the bottom of the fuel rods
- protective bottom grid with longer fuel rod end-plugs
- fuel rods positioned on the bottom nozzle
- shorter fuel assembly
- a quick release top nozzle

* The official name of the licensee is Duke Energy Corporation, as is stated in the Catawba and McGuire operating licenses. "Duke Power Company" is a component of Duke Energy Corporation; however, for historical reasons, the licensee used "Duke Energy Corporation" and "Duke Power Company" interchangeably. This safety evaluation follows the licensee's practice.

The first three design features listed above were licensed via the Wolf Creek Fuel design (Ref. 6) using the NRC-approved Westinghouse Fuel Criteria Evaluation Process (Ref. 7). The next four features are included to help mitigate debris failures and incomplete rod insertion. The licensee states that these four features will be evaluated using the 10 CFR 50.59 process. The quick release top nozzle design is similar to the Reconstitutable Top Nozzle design with modifications for easier removal. This design will be licensed by Westinghouse using the fuel criteria evaluation process.

2.0 EVALUATION

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Topical report DPC-NE-2009 provides general information of the RFA design, and describes methodologies to be used for reload design analyses to support the licensing basis for the use of the RFA design in the McGuire and Catawba reload cores. These methodologies include DEC's fuel rod mechanical reload analysis methodology, and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The report does not provide the analyses of the core design, thermal-hydraulics and transients and accidents associated with the RFA design. Therefore, this safety evaluation will only address the acceptability of the methodologies described in DPC-NE-2009 for referencing in the analyses for operations with the reactor cores having a mix of Mark-BW and RFA fuel design or a full core of RFA design.

2.1 Fuel Rod Analysis Methodology

During transition periods, the reactor cores in the McGuire and Catawba plants will have both the FCF Mark-BW fuel and the Westinghouse RFA fuel. Section 4 of DPC-NE-2009 describes the fuel rod mechanical reload analysis methodology for the RFA design. While the fuel rod mechanical analyses for Mark-BW fuel will continue to be performed using the licensee's methodology described in DPC-NE-2008P-A (Ref. 8), the Westinghouse RFA fuel thermal-mechanical analyses will be performed using the NRC-approved Westinghouse fuel performance code, PAD 3.4 Code (Ref. 9). The fuel rod design bases for the RFA design are identical to those described in WCAP-12610-P-A (Ref. 5) for the VANTAGE+ fuel.

The staff's review of fuel rod analysis methodology was performed with technical assistance provided by Pacific Northwest National Laboratory (PNNL). PNNL's review findings and conclusion, with which the staff concurs, are described in its technical evaluation report (attached to this safety evaluation). Thus, the staff has found that the DEC design limits and thermal-mechanical analysis methodologies discussed in Section 4.0 of DPC-NE-2009 are acceptable for application by DEC to the RFA fuel design up to the currently approved rod average burnup limit of 62 Gwd/MTU.

2.2 Reload Core Design Methodology

For the RFA design, the core model, core operational imbalance limits, and key core physics parameters used to confirm the acceptability of Updated Final Safety Analysis Report (UFSAR) Chapter 15 safety analyses of transients and accidents will be developed with the methodologies described in DPC-NE-1004-A (Ref. 10), DPC-NE-2011P-A (Ref. 11), DPC-NF-2010A (Ref 12), and DPC-NE-3001-PA (Ref 13). DPC-NE-2011P-A describes the nuclear design methodology for core operating limits of McGuire and Catawba plants.

DPC-NF-2010A describes McGuire and Catawba nuclear physics methodology using two-dimensional PDQ07 and 3-D EPRI-NODE-P models as reactor simulators. DPC-NE-1004A describes an alternative methodology for calculating nuclear physics data using the CASMO-3 fuel assembly depletion code and the SIMULATE-3P 3-D core simulator code for steady-state core physics calculations, substituting for CASMO-2, PDQ07 and EPRI-NODE-P used in DPC-NE-2010A. DPC-NE-3001-PA describes the methodologies, which expand on the reload design methods of DPC-NF-2010A, for systematically verifying that key physics parameters calculated for a reload core, such as control rod worth, reactivity coefficients, and kinetics parameters, are bounded by values assumed in the Chapter 15 licensing analyses. These topical reports have been approved for performing reload analyses for the B&W 177-assembly and/or Westinghouse 193-assembly cores, subject to the conditions specified in the staff's safety evaluations. Because of the similarity between the RFA design and the Mark-BW fuel design with respect to the dimensional characteristics of the fuel pellet, fuel rod and cladding, as well as nuclear characteristics, as shown in Table 2-1 of DPC-NE-2009, the staff concludes that these approved methodologies and core models currently employed in reload design analyses for McGuire and Catawba can be used to perform transition and full-core analyses of the RFA design.

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Section 3.2 of DPC-NE-2009 states that conceptual transition core designs using the RFA design have been evaluated and results show that current reload limits remain bounding with respect to key physics parameters. As described in DPC's response to a staff's question (Question 1, Ref 14), the conceptual RFA transition core designs were evaluated for the effects of partial and full cores using NRC-approved codes and methods to determine the acceptability of the current licensing bases transient analyses. Key safety parameters, such as Doppler temperature coefficients, moderator temperature coefficients, control bank worth, individual rod worths, boron concentrations, differential boron worths and kinetics data, were calculated for the conceptual core designs and compared against reference values assumed in the UFSAR Chapter 15 accident analyses. The evaluation demonstrated the expected neutronic similarities between reactor cores loaded with RFA fuel and with Mark-BW fuel, and the acceptability of key safety parameters assumed in the Chapter 15 accident analyses. Key physics parameters are calculated for each reload core and each new core design. If a key physics parameters is not bounded by the reference value in the UFSAR accident analyses, the affected accidents will be re-analyzed using the new key physics parameter, or the core will be re-designed to produce an acceptable result. The staff agrees that this is acceptable approach.

The safety evaluation for DPC-NE-1004-A requires additional code validation to ensure that the methodology and nuclear uncertainties remain appropriate for application of CASMO-3 and SIMULATE-3P to fuel designs that differ significantly from those included in the topical report data base. Though the RFA design is not expected to change the magnitude of the nuclear uncertainty factors in DPC-NE-1004, the use of zirconium diboride integral fuel burnable absorber (IFBA) in the RFA is a design change from the burnable absorber types modeled in DEC's current benchmarking data base. DEC has re-evaluated and confirmed the nuclear uncertainties in DPC-NE-1004 to be bounding. This is done by explicitly modeling Sequoyah Unit 2, Cycles 5, 6, and 7, and by performing statistical analysis of the nuclear uncertainty factors. These cores were chosen because they are very similar to McGuire and Catawba and contained both IFBA and wet annular burnable absorber (WABA) fuel. The results, listed in Table 3-1 of DPC-NE-2009, showed that the current licensed nuclear uncertainty factors for the

$F_{\Delta H}$, F_Z, and F_Q bound those for the Westinghouse fuel with IFBA and/or WABA burnable absorbers. Boron concentrations, rod worth, and isothermal temperature coefficients were also predicted and found to agree well with the measured data. In response to a staff's question (Question 2, Ref. 14) regarding the applicability of the analysis of the Sequoyah core to the McGuire and Catawba Core, DEC provided comparisons of the analysis results and the measured data of the Sequoyah cores, and a list of the differences between the Westinghouse Vantage-5H fuel design used in Sequoyah and the RFA fuel design. The differences are primarily mechanical and do not impact the nuclear performance of the fuel assembly. Design features that do impact the neutronics (i.e., mid-span mixing grids) are specifically accounted for in the nuclear models. Therefore, the results and conclusions reached based on the analysis of Sequoyah core designs are applicable to the RFA fuel design. In addition, the licensee performed a 10 CFR 50.59 evaluation for unreviewed safety question (USQ). Results are as described in response to Question 2c of Ref..14, which demonstrates that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties are applicable to the RFA design. Therefore, DPC-NE-1004A nuclear physics calculation methodology is applicable to the RFA design.

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In all nuclear design analyses, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores. The mixed core model for nuclear design analyses and the use of fuel-specific limits, described in response to a staff's question (Question 3, Ref. 14), are based on the same methodology that is used to setup a nuclear model for a reactor core containing a single fuel type. When establishing operating and reactor protection system limits (i.e., LOCA linear heat rate limit, departure from nucleate boiling (DNB), central fuel melt, transient strain), the fuel-specific limits of a conservative overlay of the limits are used. The staff concludes that the nuclear design analyses for the transition cores are acceptable.

2.3 Thermal-Hydraulic Analysis

Section 5 of DPC-NE-2009 describes the thermal-hydraulic analysis methodologies to be used for the RFA design. The thermal-hydraulic analyses for the existing Mark-BW fuel design are performed with NRC approved methodology using VIPRE-01 core thermal-hydraulic code (Ref. 15), the BWU-Z critical heat flux (CHF) correlation (Ref. 16), and the thermal-hydraulic statistical core design methodology described in DPC-NE-2004P-A (Ref. 17) and DPC-NE-2005P-A (Ref. 18). As discussed in the ensuing sections of this report, these same methodologies will be used for the analyses of the RFA design with the exception that (1) the WRB-2M CHF correlation (Ref. 19) will be used in place of the BWU-Z correlation, and (2) the EPRI bulk void fraction model will be used in place of the Zuber-Findlay model.

2.3.1 VIPRE-01 Core Thermal Hydraulic Code:

The core thermal hydraulic analysis methodology using the VIPRE-01 code for McGuire and Catawba licensing calculations is described in DPC-NE-2004P-A. The VIPRE-01 models, which have been approved for the Mark-BW fuel, are also applicable to the RFA design with appropriate input of fuel geometry and form loss coefficients consistent with the RFA design. The reference pin power distribution based on an enthalpy rise factor, $F_{\Delta H}^N$, of 1.60 peak pin from DPC-NE-2004P-A will continue to be used to analyze the RFA design.

VIPRE-01 contains various void-quality relation models for two-phase flow calculation, in addition to the homogeneous equilibrium model. Either the Levy model or the EPRI model can be chosen for subcooled boiling, and the Zuber-Findlay or EPRI void models for bulk boiling. The combination of Levy subcooled boiling correlation and Zuber-Findlay bulk boiling model gives reasonable results for void fraction. This combination is currently used for McGuire/Catawba cores with the Mark-BW fuel. However, the Zuber-Findlay correlation is applicable only to qualities below approximately 0.7, and there is a discontinuity at a quality of 1.0. The licensee proposes to replace this combination with the combination of EPRI subcooled and bulk void models. The use of the EPRI bulk void model, which is essentially the same as the Zuber-Findlay model except for the equation used to calculate the drift velocity, is to eliminate a discontinuity at qualities about 1.0. Also, the use of the EPRI subcooled void model is for overall model compatibility to have the EPRI models to cover the full range of void fraction required for performing departure-from-nucleate-boiling calculation. To evaluate the impact of these model changes, the licensee performed an analysis of 51 RFA CHF test data points using both Levy/Zuber-Findlay and EPRI models in VIPRE-01. The results show a negligible 0.1 percent difference in the minimum departure-from-nucleate-boiling ratios (DNBRs). Therefore, the staff finds that the use of the EPRI subcooled and bulk void correlations for the analysis of the RFA design is acceptable. The acceptability of this revision remains subject to the limitations set forth in the safety evaluation on VIPRE-01 (EPRI NP-2511-CCM-A), DPC-NE-2004P-A and attendant revisions.

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2.3.2 Critical Heat Flux (CHF) Correlation:

The licensee stated that the WRB-2M CHF correlation, described in the Westinghouse topical report WCAP-15025-P-A (Ref. 19), will be used for the RFA design. The WRB-2M correlation was developed by Westinghouse for application to new fuel designs such as the Modified Vantage 5H and Modified Vantage 5H/IFM. The WRB-2M correlation was programmed into the Westinghouse thermal hydraulic code THINC-IV, or the VIPRE-01 thermal-hydraulic code for the calculation of the local conditions within the rod bundles. The staff has reviewed and approved the WRB-2M correlation with both THINC-IV and VIPRE-01 codes as described in References 20 and 21. The WRB-2M correlation is also applicable to the RFA design because of its similarity to the Vantage 5H fuel design. The staff concludes DEC's use of the WRB-2M along with VIPRE-01 in the DNBR calculations for the RFA design to be acceptable within the ranges of applicability of important thermal hydraulic parameters specified in the staff's safety evaluation on WCAP-15025-P-A (Ref. 20).

2.3.3 Thermal-Hydraulic Statistical Core Design Methodology

The thermal-hydraulic analysis for the RFA design will be performed with the statistical core design (SCD) analysis method described in DPC-NE-2005P-A, Rev. 1 (Ref. 18). The SCD analysis technique differs from the deterministic thermal hydraulic method in that the effects on the DNB limit of the uncertainties of key parameters are treated statistically. The SCD methodology involves selection of key DNBR parameters, determination of their associated uncertainties, and propagation of uncertainties and their impacts to determine a statistical DNBR limit that provides an assurance with 95% probability at 95% confidence level that DNB will not occur when the nominal values of the key parameters are input in the safety analysis. The SCD methodology described in DPC-NE-2005P-A is identical to the SCD methodology described in

DPC-NE-2004P-A (Ref. 17) with the exception that the intermediate step of using a response surface model to evaluate the impact of uncertainties of key DNBR parameters about a statepoint is eliminated and replaced with the VIPRE-01 code to directly calculate the DNBR values for each set of reactor conditions. The staff has approved the SCD methodology with restrictions that: (1) its use of specific uncertainties and distributions will be justified on a plant-specific basis, and its selection of statepoints used for generating the statistical design limit will be justified to be appropriate; and (2) only the single, most conservative DNBR limit of two limits proposed by DPC for separate axial power distribution regions is acceptable. The licensee subsequently submitted Appendix C to DPC-NE-2005P-A containing the plant-specific data and limits with Mark-BW 17x17 type fuel using the BWU-Z CHF correlation, the VIPRE-01 thermal-hydraulic computer code, and DEC SCD methodology to support McGuire and Catawba reload analyses. The staff previously found the BWU-Z correlation and the statistical DNBR design limit to be acceptable for the Mark-BW 17x17 fuel (Ref. 16).

Table 5.3 of DPC-NE-2009 provides McGuire/Catawba plant-specific data on the uncertainties and distributions, as well as the justifications, of the SCD parameters, the WRB-2M CHF correlation, and the VIPRE-01 code/model. Table 5-4 provides the McGuire/Catawba statepoint statistical results with the WRB-2M CHF correlation for the RFA core. The statistical design limit of DNBR of 1.30 for the RFA core is chosen to bound the all statistical DNBRs. The staff finds them acceptable for the RFA design.

2.3.4 Transition Cores:

The licensee stated that for the operation with transitional mixed cores having both the Mark-BW fuel and RFA designs, the impact on the thermal hydraulic behavior of the geometric and hydraulic differences between these two fuel designs will be evaluated with an 8-channel core model. This is done by placing the RFA design in the channels representing the limiting hot assembly and the Mark-BW fuel assemblies in the eighth channel representing the rest of the assemblies. The transition core analysis models each fuel type in their respective locations with correct geometry and the form loss coefficients. A transition core DNBR penalty is determined for the RFA design, and a conservative DNBR penalty is applied for all DNBR analyses for the RFA/Mark-BW transition cores.

To determine the transition mixed core DNBR penalty, the licensee has re-analyzed the most limiting full core statepoint used in the SCD analysis using the 8-channel transition core model. The result of the transition core DNBR showed an increase of statistical DNBR by less than 0.2%, and the DNBR value is still less than the statistical design limit of 1.30 for the full core of RFA design with the WRB-2M CHF correlation. Therefore, the staff concludes that the statistical design limit of 1.30 can be used for both transition and full core analyses.

2.4 <u>UFSAR Accident Analyses</u>

To support the operation of transitional Mark-BW/RFA mixed core and full RFA cores, the UFSAR Chapter 15 transients and accidents analyses will be performed. The LOCA analyses will be performed by Westinghouse using approved LOCA evaluation models. Non-LOCA transients and accidents will be performed by the licensee using previously approved methodologies.

2.4.1 LOCA Analyses:

Westinghouse will perform the large- and small-break LOCA analyses for the operations of transition and full-core of the RFA design using approved versions of the Westinghouse Appendix K LOCA evaluation models (EM). The small-break LOCA EM (Ref. 22, 23) includes the NOTRUMP code for the reactor coolant system transient depressurization, and the LOCTA-IV code for the peak cladding temperature calculation. The large-break LOCA EM (Ref. 24) includes BASH and other interfacing codes such as SATAN-VI, REFILL, and LOCBART, for various phases. For operation of the transient Mark-BW/RFA cores, explicit analyses will be performed simulating the cross-flow effects due to any hydraulic mismatch between the Mark-BW and the RFA design. The licensee stated that if it determined that a transition core penalty is required during the mixed core cycles, it will be applied as an adder to the LOCA results for a full core of the RFA design. Since the Westinghouse LOCA EMs, both the large- and small-break, are approved methodologies for PWR fuel designs, the staff concludes they are acceptable for performing LOCA analyses for the RFA design.

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2.4.2 Non-LOCA Transient and Accident Analyses:

The safety analyses of McGuire and Catawba UFSAR Chapter 15 non-LOCA transients and accidents are performed with the RETRAN-02 system transient code and the VIPRE-01 core thermal-hydraulic code. The non-LOCA transient analysis methodologies are described in several topical reports. DPC-NE-3002-A, Rev. 1 (Ref. 25) describes the system transient analysis methodology including the RETRAN model nodalization, initial and boundary conditions, and input assumptions regarding control, protection, and safeguard system functions used in the safety analyses of all Chapter 15 non-LOCA transients and accidents, except for those involving significant asymmetric core power peaking. DPC-NE-3001-PA describes the methodologies for systematically confirming that reload key physics parameters are bounded by values assumed in the Chapter 15 safety analyses, and for analyses of the events of control rod ejection, steam line break, and dropped rod, which involve significant asymmetric core power peaking and require evaluation of multi-dimensional simulations of the core responses. DPC-NE-2004P-A and DPC-NE-2005P-A describe the procedure used to apply the VIPRE-01 code for the reactor core thermal-hydraulic analyses and the SCD methodologies for the derivation of the statistical DNBR limit. DPC-NE-3000-PA (Ref. 26) documents the development of thermal-hydraulic simulation models using RETRAN-02 and VIPRE-01 codes, including detailed descriptions of the plant nodalizations, control system models, code models, and the selected code options for McGuire and Catawba plants.

These methodologies have been previously approved by NRC for the analyses of non-LOCA transients and accidents for McGuire and Catawba with the Mark-BW fuel design. A change of reactor core fuel from Mark-BW to the RFA design does not affect the conclusion of the analytical capabilities of RETRAN-02 and VIPRE-01, except for the need to change the inputs to reflect the RFA design in the safety analyses. The licensee performed a review of DPC-NE-3000-PA and identified the necessary changes in the existing transient analyses methods for performance of safety analyses in support of the RFA design. Minor changes are required to the volume, and associated junction and heat conductor calculations in the reactor core region of the RETRAN primary system nodalization model to reflect the dimensional changes to the RFA design. Input changes to the VIPRE model are required in core thermal hydraulic analysis to reflect the RFA design geometry and form loss coefficients. In addition, as discussed in Sections 2.3.2 and 2.4.3, respectively, of this safety evaluation, the WRB-2M CHF correlation will be used for the DNBR calculation, and the SIMULATE-3K code will be used in place of SIMULATE-3P for the nuclear portion of the control rod ejection accident analysis. The staff concludes the non-LOCA safety analysis methodologies are acceptable for the RFA design.

2.4.3 Rod Ejection Accident Analysis Using SIMULATE-3K:

The rod ejection accident (REA) analysis methodology described in DPC-NE-3001-PA includes the use of three-dimensional space-time transient neutronics nodal code ARROTTA (Ref. 27) to perform the nuclear analysis portion of transient response; the VIPRE-01 code to model the core thermal response including peak fuel enthalpy, a core-wide DNBR evaluation, and transient core coolant expansion; and the RETRAN-02 code to simulate the reactor coolant system pressure response to the core power excursion. This methodology will continue to be used for the REA analysis except for the use of the SIMULATE-3K code (Ref. 28) to replace ARROTTA to perform the nuclear analysis of the response of the reactor core to the rapid reactivity insertion resulting from a control rod being ejected out of the core.

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Section 6.6 of DPC-NE-2009 describes the REA analysis methodology using SIMULATE-3K, including a brief description of the code and models, code verification and benchmark, and the REA analysis application of SIMULATE-3K. SIMULATE-3K is a three-dimensional transient neutronic version of the NRC approved SIMULATE-3P computer code (Ref. 29), and uses the same neutron cross section library. It uses a fully-implicit time integration of the neutron flux, delayed neutron precursors, and heat conduction models. The average beta for the time-varying neutron flux is determined by performing a calculation of the adjoint flux solution. The code user has the option of running the code with a fixed time step or a variable time step depending on the sensitivity to changes in the neutronics. The SIMULATE-3K code has incorporated additional capability to model reactor trips at user-specified times in the transient or following a specified excore detector response, which allows the user to specify the response of individual detectors as required to initiate the trip, as well as the time delay prior to release of the control rods based on the excore detector response model. The code also permits the user input to control the velocity of the control rod movement, providing a different perspective for each velocity chosen.

The SIMULATE-3K code vendor, Studsvik of America, Inc., had performed the code verification and validation during its development to verify correctness of the coding, and to validate the applicability of the code to specified analyses and ensure compatibility with existing methodology. The validation included benchmarks of the fuel conduction and thermal hydraulic models, the transient neutronics model, and the coupled performance of the transient neutronics and thermal-hydraulic models. The fuel and thermal hydraulic models were validated against the TRAC code, while the neutronic model was benchmarked against the solutions of the industry standard light water reactor problems generated by QUANDRY, NEM, and CUBBOX (Ref. 30,31, 32). Benchmarking of the coupled performance of the thermal hydraulic and transient neutronics models was carried out against the results from a standard NEACRP [Nuclear Energy Agency Control Rod Problem] rod ejection problem to the PANTHER code (Ref. 33). Steady state comparison of S3K was performed against the NRC approved COSMO-3/SIMULATE-3P. In addition, DPC performed comparisons of the SIMULATE-3K and

ARROTTA calculations for the reference REA analysis for the Oconee Nuclear Station, and showed very good agreement for core power versus time for the ejection occurring at the end-of-cycle from the maximum allowable power level with 3 and 4 RCPs operating and from both beginning-of-cycle and end-of-cycle at hot zero power and hot full power conditions. These SIMULATE-3K validation benchmarks were presented in DPC-NE-3005-P (Ref. 34), which the staff has reviewed for approval of using SIMULATE-3K for the analysis of the REA for the Oconee plants.

Section 6.6.1.3.3 of DPC-NE-2009 provides additional benchmark of SIMULATE-3K by comparing the SIMULATE-3K and ARROTTA calculations for the reference REA analyses performed at beginning of life (BOC) and end of life (EOC), HFP and HZP conditions for McGuire and Catawba Nuclear Stations. The reference core used in the benchmark calculations was a hypothetical Catawba 1 Cycle 15 core, which represents typical fuel management strategies currently being developed for reload core designs at McGuire and Catawba. The comparison between the SIMULATE-3K and ARROTTA calculations of the core power level and nodal power distribution as functions of time during the REA transient demonstrate the acceptability of the physical and numerical models of the SIMULATE-3K for application in the REA analyses for McGuire and Catawba Nuclear Station.

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Section 6.6.2.2 of DPC-NE-2009 describes the use of the SIMULATE-3K code to perform license analysis of the design basis REA. The basic methodology as described in DPC-NE-3001PA remains unchanged with the exception of minor differences between SIMULATE-3K and ARROTTA. The core power levels and nodal power distributions calculated by SIMULATE-3K are used VIPRE to determine the fuel enthalpy, the percentage of fuel pins exceeding the DNB limit, and the coolant expansion rate. All inputs to VIPRE, once supplied by the NRC approved code ARROTTA, are now supplied by SIMULATE-3K.

In the SIMULATE-3K nuclear analysis of an REA, a fuel assembly is typically geometrically modeled by several radial nodes. Axial nodalization and the number of nodes are chosen to accurately describe the axial characteristics of the fuel. For current fuel designs, a typical axial nodalization of 24 equal length fuel nodes in the axial direction is used. SIMULATE-3K explicitly calculates neutron leakage from the core by use of reflector nodes in the radial direction beyond the fuel region and in the axial direction above and below the fuel column stack. The fuel and reflector cross sections are developed in accordance with the methodology described in the approved topical report DPC-NE-1004A for SIMULATE-3P.

The SIMULATE-3K REA analysis is performed at four statepoints: BOC and EOC at HZP and HFP conditions for the determination of three-dimensional steady-state and transient power distributions, as well as individual pin powers. Conservative input parameters are used to ensure that the rod ejection analysis produces limiting results that are expected to bound future reload cycles. Sections 6.6.2.2.1 and 6.6.2.2.2 describe the methods to ensure conservatism in the analysis of transient response by increasing the fission cross sections in the ejected rod locations and in each assembly and by applying the "factors of conservatism" to the reactivity feedback for moderator and fuel temperatures, control rod worths for withdrawal and insertion, effective delayed neuron, and ejected rod worth, etc. In response to a staff question (No. 9, Ref. 14), the licensee provided a description of the method of determining the "factors of conservatism." The staff has reviewed the overall SIMULATE-3K methodology, and found it to be acceptable for application to the REA analyses for McGuire and Catawba.

2.4.4 Compliance with Safety Evaluation Conditions:

As discussed above, licensing analyses of reload cores with the RFA design use the methodologies described in various topical reports for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current McGuire/Catawba cores. These methodologies may have inherent limitations, or conditions or restrictions imposed by the associated NRC safety evaluations in their applications. The acceptability of the licensing analyses is subject to the application being within the limitations of the methodologies used, and the conditions or restrictions imposed in the respective safety evaluations. In response to a staff question regarding the resolutions of these limitations, conditions, and restrictions in the RFA reload safety analyses, the licensee provided (Response to Question 11, Ref. 14) a list of restrictions imposed by NRC safety evaluations and the corresponding resolutions in the application of the licensee's methodologies used for the safety analyses of the non-LOCA transients and accidents. In addition, for the LOCA analyses to be performed by Westinghouse, the licensee provided a Westinghouse response (Ref. 35) regarding the safety evaluation restrictions and corresponding compliance for the 1985 SBLOCA Evaluation Model with NOTRUMP and the 1981 Evaluation Model with BASH. The resolutions or compliance with the conditions or restrictions provided in these responses provide guidance for the licensee referencing DPC-NE-2009 in the RFA reload licensing analyses. The staff concludes that the safety evaluation conditions have been properly addressed.

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2.5 Fuel Assembly Repair and Reconstitution

Section 7.0 of DPC-NE-2009 describes the evaluation of the reconstitution or repair of fuel assemblies having failed fuel rods during refueling outages in an effort to achieve the zero fuel defect goal during cycle operation. The primary replacement candidate for use in reconstitution of failed fuel rods is a fuel rod that contains pellets of natural uranium dioxide, but solid filler rods made of stainless steel, zircaloy, or ZIRLO would be used if local grid structural damage exists. The reconstitution of the RFA assembly with filler rods will be analyzed with NRC-approved methodology and guidelines described in DPC-NE-2007P-A (Ref. 36), along with other licensed codes and correlations, to ensure acceptable nuclear, mechanical, and thermal-hydraulic performance of reconstituted fuel assemblies.

For a reload core using reconstituted Westinghouse fuel, Westinghouse has reviewed the effects of the reconstituted fuel with the criteria specified in Standard Review Plan 4.2 and determined that the only fuel assembly mechanical criteria impacted by reconstitution are fuel assembly holddown force and assembly structural response to seismic/LOCA loads. Westinghouse has evaluated these effects on the LOCA analyses using the approved methodology WCAP-13060-P-A (Ref. 37), and concluded that the reconstituted fuel assembly designs are acceptable for both normal and faulted condition operations.

2.6 Technical Specifications Changes

The licensee's July 22 and October 22, 1998, letters (Ref. 1 and 2) proposed changes to the Technical Specifications with the technical justifications for these changes described in Chapter 8 of DPC-NE-2009. These proposed TS changes are described and evaluated in the following sections.

2.6.1 Proposed Change to TS Figure 2.1.1-1

The licensee proposed to modify Figure 2.1.1-1, "Reactor Core Safety Limits - Four Loops in Operation," by deleting the 2455 psia safety limit line, which is the current upper bound pressure allowed for power operation.

The 2455 psia bounding pressure is based on the pressure range of the CHF correlation used in DNBR analyses of the Mark-BW fuel. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2455 psia safety limit line is deleted, and the remaining safety limit lines with 2400 psia as the upper bound safety limit line are within the range of the CHF correlations for the Mark-BW and RFA fuel designs. As described in its response to a staff's question (No. 12, Ref. 14), the licensee has performed an evaluation to ensure the remaining safety limit lines of Figure 2.1.1-1, which were based on the CHF correlation for the Mark-BW fuel design and the hot leg boiling limit, bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design. Both the full RFA core and the transition RFA/Mark-BW cores were evaluated to ensure that the established limits were conservative. The MDNBR values were greater than the design DNBR limit for all the cases in both evaluation. Therefore, the safety limit lines in Figure 2.1.1-1, with the deletion of the 2455 psia safety limit line, are acceptable.

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2.6.2 Proposed Changes to Surveillance Requirements 3.2.1.2, 3.2.1.3, and 3.2.2.2

TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor F_q (x,y,z) and the enthalpy rise hot channel factor $F_{\Delta h}$ (x,y) to be measured periodically (once within 12 hours after achieving equilibrium conditions after a power change exceeding 10% rated thermal power, and every 31 effective full power days thereafter) using the incore detector system to ensure the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limit are not violated. To avoid the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the F_q (x,y,z) or $F_{\Delta h}$ (x,y) has decreased since the previous surveillance. The 2% margin-decrease penalty was based on the current reload cores.

For the reactor core containing the RFA fuel design with integral burnable absorbers, larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. The licensee proposed to remove the 2% penalty value from these SRs and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide typical values for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors. The actual values for the transitional core can not be provided until the final design for the core is complete. In response to a staff's question (No. 13, Ref. 14), the licensee provided the methodology for calculating the burnup-dependent penalty factors. In addition, Technical Specification 5.6.5 will reference topical report DPC-NE-2009, which includes this response to the staff's question for the approved methodology used to calculate these penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin-decrease penalty factors in the COLR acceptable.

2.6.3 Proposed Change to TS 4.2.1

TS 4.2.1, "Fuel Assembles," which specifies the design features for fuel assemblies, will be revised to add ZIRLO cladding to the fuel assembly description.

2.6.4 Proposed Changes to Section 5.6.5b

By a letter dated May 6, 1999 (Ref. 38), the licensee expanded the original amendment request by proposing more changes in Section 5.6.5. The section lists all the topical reports previously approved by the staff. Thus these proposed changes are administrative or editorial. The staff finds them all acceptable as follows:

> WCAP-10216P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification" -- This is deleted since it had been previously replaced by Item 5 (re-numbered Item 4), DPC-NE-2011P-A.
>
> BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants" -- The dates of the various staff safety evaluations have been updated.
>
> DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology" -- The Revision number has been changed from "2" to "3". The staff's safety evaluation date is also updated.
>
> DPC-NE-3000P-A, "Thermal-Hydraulic Transient Analysis Methodology" -- The Revision number is changed from "1" to "2". The staff's safety evaluation date is also updated.
>
> DPC-NE-2001P-A, "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel" -- This is deleted, and is replaced by DPC-NE-2008P-A.
>
> BAW-10183P-A, "Fuel Rod Gas Pressure Criterion" -- This is deleted. DPC-NE-2008P-A references this report, and therefore there is no need for an individual listing.
>
> WCAP-10054P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code" -- This report is applicable to the Westinghouse fuel.
>
> DPC-NE-2009P-A, "Westinghouse Fuel Transition Report" -- This report has been evaluated in the above sections of this safety evaluation and found acceptable.

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2.6.5 Proposed Changes to the Technical Specifications Bases Document

The TS Bases is a licensee-controlled document, and is not part of the Technical Specifications (10 CFR 50.36(a)). However, the staff reviewed the licensee's proposed changes as supplemental information for the TS changes evaluated above. The Bases sections for SR 3.2.1.2, 3.2.1.3 and 3.2.2.2 will be revised to reflect the corresponding TS changes. The staff finds the proposed changes to the Bases acceptable.

3.0 REVIEW SUMMARY OF TOPICAL REPORT

The staff has reviewed the licensee's Topical Report DPC-NE-2009P and found it acceptable for referencing for analysis for reloads with Westinghouse RFA design. The topical report references many topical reports which provide methodologies for various aspects of the RFA reload licensing analyses. Acceptability of DPC-NE-2009P remains subject to the limitations set forth in the SERs on these topical reports.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, South Carolina State official Mr. Virgil Autrey was notified of the proposed issuance of the amendments to Catawba. The State official had no comments.

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5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 64108, dated November 18, 1998; 69 FR _____, dated May __, 1999) . Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date:

4 REFERENCES

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2. Letter, M. S. Tuckman (Duke Energy Corporation) to NRC, "Duke Energy Corporation, McGuire Nuclear Station, Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Supplement to License Amendment Request for Revisions to Improved Technical Specifications for Implementation of Westinghouse Fuel as Described in Topical Report Described in Topical Report DPC-NE-2009/DPC-NE-2009P, Duke Power Company Westinghouse Fuel Transition Report," October 22, 1998

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7. Davison, S. L., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-P-A, October 1994

8. "Duke Power Company Fuel Rod Mechanical Reload Analysis Methodology Using TACO3," DPC-NE-2008P-A, April 1995

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17. DPC-NE-2004P-A, Rev. 1, "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," February, 1997

18. DPC-NE-2005P-A, Rev. 1, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," November, 1996

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